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Final Safety Evaluation Report Related to the Certification of the System 80 + Design Docket No. 52-002

Chapters 1-14

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ABSTRACT

This final safety evaluation report (FSER) documents the technical review of the System 80+ standard design by the U.S. Nuclear Regulatory Commission (NRC) staff. The application for the System 80+ design was initially submitted by Combustion Engineering, Inc., now Asea Brown Boveri-Combustion Engineering (ABB-CE), in accordance with the procedures of Appendix O of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). Later ABB-CE requested that its application be considered as an application for design approval and subsequent design certification pursuant to 10 CFR § 52.45.

System 80+ is a pressurized water reactor with a rated power of 3914 megawatts thermal (MWt) and a design power of 3992 MWt at which accidents are analyzed. Many features of the System 80+ design are similar to those of ABB-CE's System 80 design from which it evolved. The staff approved the System 80 design in NUREG-0852, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply

Reference System CESSAR System 80," dated November 1981, and in supplements thereto. Unique features of the System 80+ design include: a large spherical, steel containment; an in-containment refueling water storage tank; a reactor cavity flooding system, hydrogen ignitors, and a safety depressurization system for severe accident mitigation; a combustion gas turbine for an alternate ac source; and an advanced digitally based control room.

On the basis of its evaluation and independent analyses, the NRC staff concludes that ABB-CE's application for design certification meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant to the System 80+ standard design. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR § 52.53 is provided in Appendix E. A final design approval, issued on the basis of this FSER, does not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, the Atomic Safety and Licensing Board, and other presiding officers, in any proceeding pursuant to Subpart G of 10 CFR Part 2.

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On March 30, 1989, Combustion Engineering, Inc. tendered its application for certification of the System 80+ standard design with the U.S. Nuclear Regulatory Commission (hereinafter referred to as the NRC, the Commission, or the staff). The submittal was made in accordance with Appendix O, "Standardization of Design: Staff Review of Standard Designs," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). Combustion Engineering, Inc., subsequently stated in a letter dated August 21, 1989, that its application may be considered as an application for design approval and subsequent design certification pursuant to 10 CFR 52.45. The application was docketed on May 1, 1991, and assigned Docket No. 52-002. Correspondence relating to the application prior to this date was also addressed to Docket No. 50-470 and Project No. 675. By letter dated May 26, 1992, Combustion Engineering, Inc. notified the NRC that it is a wholly owned subsidiary of Asea Brown Boveri, Inc., and the appropriate abbreviation for the company is ABB-CE. Therefore, the staff refers to Combustion Engineering, Inc. as ABB-CE throughout this report.

The NRC's licensing project managers assigned to the System 80+ standard design review are Mr. Thomas Wambach, Mr. Michael Franovich, and Mr. Stewart Magruder. They may be reached by calling (301) 415-7000 or by writing to the Office of Nuclear Reactor Regulation, Mail stop O-11 H3, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001.

ABB-CE's application, the Combustion Engineering Standard Safety Analysis Report Design Certification (CESSAR-DC), describing the design of the facility, was originally submitted on March 30, 1989. Subsequently, ABB-CE supplemented the information in CESSAR-DC through an amendment process. The most recent amendment, Amendment W, was submitted to the Commission on June 20, 1994. ABB-CE also submitted the System 80+ certified design material (CDM) by letter dated December 31, 1993. Subsequently, ABB-CE submitted a revision incorporating resolution of NRC comments by letter dated June 20, 1994. The CDM contains the Tier 1 design information that ABB-CE proposes to have certified during the System 80+ design certification rulemaking. The application, the CDM, and all CESSAR-DC amendments are available for public inspection at the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. 20037. A summary of the System 80+ standard design appears in Sections 1.2 and 1.3 of this report. Section 1.4 identifies agents and contractors.

This final safety evaluation report (FSER) summarizes the staff's safety review of the System 80+ design against the requirements of Subpart B of 10 CFR Part 52 (design certification) and delineates the scope of the technical details considered in evaluating the proposed design. The principal matters of the staff's review are discussed in Section 1.5. This report conforms to the Commission's Policy Statement on metrication. Therefore, all units are expressed as metric followed by English units in parentheses. However, in response to a letter by ABB-CE dated April 2, 1993, the Commission exempted design certification of System 80+ from the Commission's policy on metrication. Therefore, the CESSAR-DC and CDM use English units.

An advance FSER was issued on March 8, 1994. The purpose of this advance FSER was to document resolution of the open and confirmatory items identified in the draft safety evaluation report (DSER) for the System 80+, draft NUREG-1462, which was issued on October 1, 1992, and to facilitate the review by the Advisory Committee on Reactor Safeguards (ACRS) and the Office of the General Counsel. The advance FSER documents the staff's review up to and including Amendment U of CESSAR-DC.

This FSER revises the advance FSER to incorporate the results of technical editor comments, the Office of General Counsel comments, the Advisory Committee on Reactor Safeguards (ACRS) comments, and closure of the confirmatory items listed in Section 1.7 of the advance FSER. Confirmatory Item 1.1-1 involved verification by the staff that all changes committed to by ABB-CE were incorporated in Amendments V and W to CESSAR-DC. The staff has verified this and Confirmatory Item 1.1-1 is resolved. Confirmatory Item 1.1-2 involved ensuring that all information in the CESSAR-DC, final technical specifications (TS), CDM, and this FSER are consistent within each document and between documents. The staff has completed this review and Confirmatory Item 1.1-2 is resolved. Confirmatory Item 1.10-1 involved the staff verifying that the issues outside the scope of design certification and that must be addressed by an applicant for a combined license that references System 80+ are properly identified in Table 1.10-1 of CESSAR-DC. The staff completed this verification and this item is resolved. The other five confirmatory items in Section 1.7 of the advance FSER are resolved and addressed in the appropriate sections of this FSER.

Several references to ABB-CE reports are made in this report. Some of these reports contain information that has been authorized by the Commission to be exempt from public disclosure, as provided by 10 CFR 2.790. For each such report containing proprietary information, a nonproprietary version, similar in content except for the omission

Introduction and General Discussion

of the proprietary information, is provided to the NRC by ABB-CE and is also available at the NRC Public Document Room. Several references to ABB-CE reports throughout this FSER are made to the proprietary version only. The staff based its findings on the proprietary versions of these documents.

In its application, ABB-CE stated that the criteria included in the Electric Power Research Institute's (EPRI) Advanced Light Water Reactor Program would be used as a guide for the design of System 80+. The Commission had requested that the staff evaluate any differences between the vendor designs and the EPRI Utility Requirements Document (URD) in a staff requirements memorandum (SRM) dated December 15, 1989. On December 21, 1990, ABB-CE sent the staff a summary of the differences between its design and the EPRI URD. In the DSER, the staff identified an open item (DSER Open Item 1.1-1) for ABB-CE to address any System 80+ deviations from the EPRI URD. Subsequently, ABB-CE indicated that it was a principal participant in the development of the EPRI sponsored URD and continue to be involved with EPRI on changes to that document. Therefore, the design in CESSAR-DC remains consistent with the EPRI URD. The Commission designated that this response was acceptable in COMSECY-93-040, dated August 10, 1993. In a letter of January 7, 1994, ABB-CE stated that the System 80+ design was consistent with the EPRI URD. The staff finds this acceptable. On this basis, DSER Open Item 1.1-1 is resolved.

Applicants who reference the System 80+ standard design in the future for specific plants will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a combined license (COL), the staff will evaluate, for each plant-specific application that references the System 80+ design, the technical competence of the COL applicant and its contractors to manage, design, construct, and operate a nuclear power plant. The plant-specific applicants will also be required to satisfy the requirements of Subpart C of 10 CFR Part 52, and the requirements resulting from the staff's review of this standard design, including the applicable regulations and exemptions identified in Section 1.6 of this report. ABB-CE has also identified requirements for plant-specific applicants as "COL License Information." These requirements are discussed in the CESSAR-DC Section 1.10 and are listed in Table 1.10-1 of CESSAR-DC. The staff reviewed this list for completeness and finds it acceptable. Confirmatory Item 1.10-1 of the advance FSER is resolved.

A chronology of the principal actions, submittals, and amendments related to the processing of the application is given in Appendix A of this report. An index of the

staff's requests for additional information (RAIs) and the responses made by ABB-CE appears in Appendix B. A list and definition of the abbreviations used throughout this report is provided in Appendix C. Appendix D provides a list of the principal technical reviewers who evaluated CESSAR-DC. Appendix E is the letter from the ACRS to the Commission regarding the results of its review of the System 80+ design.

1.2 General Design Description

1.2.1 Scope of System 80+ Design

Section 52.47(b)(1) of 10 CFR Part 52 requires an applicant for certification of an evolutionary nuclear power plant design to submit an essentially complete design scope. Therefore, the scope of the System 80+ design must include all elements of the plant which can affect safe operation except for such site-specific elements as the service water intake structure and the ultimate heat sink.

When the DSER was issued, the staff was still reviewing the division of the structures and systems between the certified scope and the site-specific elements. This was designated as DSER Open Item 1.2-1. ABB-CE provided a discussion of the scope of the certified design in CESSAR-DC Section 1.2. A list of the structures and systems is provided in Table 1.1 of this report. CESSAR-DC Figure 1.2-1 provided the scope of the System 80+ design, and Table 1.9-1 indexed interface requirements. The staff finds this acceptable. This resolved DSER Open Item 1.2-1.

1.2.2 Summary of the System 80+ Design

The nuclear steam supply system (NSSS) consists of a pressurized-water reactor with two primary coolant loops, a pressurizer connected to one of the loops, and the auxiliary and safety systems directly related to the NSSS. The rated core thermal power of the NSSS is 3914 MWt, producing saturated steam for use in the balance of plant (BOP) steam and power conversion system. The design core power at which accidents are analyzed is 3992 MWt. The CESSAR-DC is based on a single-unit site plan, as shown in CESSAR-DC Figure 1.2-1. This power level exceeds the guidance in Regulatory Guide (RG) 1.49, "Power Levels of Nuclear Power Plants," which states that licensed power levels should be limited to a reactor core power level of 3800 MWt or less until January 1, 1979, at the earliest. The intent of this regulatory guidance was to stabilize the maximum size of nuclear plants until sufficient experience was gained with design, construction, and operation of large plants. Since RG 1.49, Revision 1 was

Table 1.1 System 80+ design certification scopeReactor Systems

Reactor coolant system
 Chemical and volume control system
 Process sampling system
 Primary sampling system
 Secondary chemistry control system
 Fuel system
 Fuel storage and handling systems

Safeguards Systems

Safety depressurization system
 Safety injection system
 Shutdown cooling system
 In-containment water storage system
 Reactor coolant gas vent system
 Emergency feedwater system
 Cavity flooding system

Steam and Power Conversion Systems

Main steam supply system
 Condensate and feedwater system
 Main condenser
 Condensate storage system
 Condensate cleanup system
 Main condenser evacuation system
 Demineralized water makeup system
 Extraction steam system
 Heater drain system
 Heater vents
 SG blowdown system
 SG wet layup recirculation system

Turbine Generator Systems

Turbine generator
 Turbine bypass system
 Turbine gland sealing system
 Turbine lube oil system
 Turbine control system
 Turbine generator cooling system

Waste Management Systems

Liquid waste management system
 Gaseous waste management system
 Solid waste management system
 Process and effluent radiation monitoring system

Containment Systems

Containment spray system
 Containment isolation system
 Containment hydrogen recombiner system
 Hydrogen mitigation system
 Containment vessel

Cooling Water Systems

Component cooling water system
 Pool cooling and purification system
 Station service water system
 Turbine building service water system
 Turbine building cooling water system
 Chilled water systems
 Essential chilled water system
 Normal chilled water system

Support Systems

Compressed air systems
 Instrument air system
 Station air system
 Breathing air system
 Compressed gas systems
 Fire protection system
 Equipment and floor drainage system
 Diesel generator systems
 Diesel generator engine air intake and exhaust system
 Diesel generator engine cooling water system
 Diesel generator engine fuel oil system
 Diesel generator engine lube oil system
 Diesel generator engine starting air system
 Control building ventilation system
 Fuel building ventilation system
 Nuclear annex ventilation system
 Reactor building subsphere ventilation system
 Diesel building ventilation system
 Annulus ventilation system
 Containment purge ventilation system
 Containment cooling and ventilation system
 Turbine building ventilation system
 Communication systems
 Lighting systems
 Normal lighting
 Security lighting
 Emergency lighting

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Table 1.1 System 80+ design certification scope (continued)

<u>Instrumentation and Control Systems</u>	<u>Onsite Power Systems (continued)</u>
Plant protection system Reactor protection system Engineered safety features actuation system Control systems not required for safety Discrete indication and alarm system Integrity monitoring system Data processing system	Dc power systems (continued) Class 1E dc power systems 125V dc and 120V ac vital instrumentation and control power system
<u>Onsite Power Systems</u>	<u>Structures</u>
Ac power systems Class 1E ac power systems Class 1E standby power supplies Non-Class 1E ac power systems Non-Class 1E alternate ac source standby power supply Dc power systems Non-Class 1E dc power systems 125V dc auxiliary control power system 208/120V ac auxiliary control power systems 250V dc auxiliary control power system	Nuclear island Reactor building Nuclear annex Control complex area Technical support center and operations support center Fuel pool area Maintenance/outage area Diesel generator rooms Main steam valve houses Chemical and volume control system area Turbine building

issued in 1973, the staff has reviewed sufficient operating experience and has determined that licensing the System 80+ at a rated power of 3914 MWt is acceptable.

The reactor core consists of 241 fuel assemblies and 93 or more control element assemblies (CEAs). Each fuel assembly is a 16 x 16 array consisting of 236 fuel and poison rods and 5 guide tubes. The fuel rods are Zircaloy tubes containing slightly enriched uranium dioxide pellets. Some also contain burnable poison. Full-strength CEAs consist of an Inconel-clad, boron carbide absorber, or a silver-indium cadmium (Ag-In-Cd) absorber, and part-strength CEAs are composed of Inconel clad tubes loaded with Inconel slugs over the full active length.

The reactor coolant system (RCS) consists of reactor coolant (borated water) and two independent loops connected in parallel to the reactor. Each loop consists of a 107 cm (42 in.) inner diameter (ID) outlet pipe, two 76 cm (30 in.) ID inlet pipes, a steam generator (SG), and two

reactor coolant pumps (RCPs). The RCS operates at a nominal pressure of 15,500 kPa (2250 psia). The system pressure is maintained by an electrically heated pressurizer that is connected to one of the loops. Each SG is a vertical U-tube heat exchanger used to transfer heat generated in the reactor core and carried by the reactor coolant to produce steam used for driving the plant turbine-generator.

The System 80+ plant protective system (PPS) consists of the reactor protective system (RPS) and the engineered safety features actuation system (ESFAS). The RPS initiates a reactor trip if the reactor reaches a prescribed safety limit. The ESFAS sends an actuation signal to the engineered safety features (ESF) systems when a fluid system or containment parameter is reached. The PPS is augmented by the alternate protection system (APS), which generates an alternate reactor trip signal and alternate feedwater actuation signal that are separate and diverse from the PPS. A separate reactor control system is

provided for startup and shutdown of the reactor, and adjustments to reactor power in response to turbine load changes. Ex-core and in-core nuclear instrumentation and the process monitoring system are provided to monitor power, flux distribution within the core, temperature, pressure, flow, and liquid level and provide inputs to the PPS and the reactor control system.

The containment vessel is a 61 m (200 ft) diameter spherical-shaped steel shell with a wall thickness of approximately 4.5 cm (1.75 in.). The containment vessel is completely enclosed by the shield building. The shield building is a reinforced concrete cylindrical building with a hemispherical dome. The annular space between the containment and the shield building is filtered and recirculated or exhausted during accident conditions by the annulus ventilation system. Space below the containment and inside the shield building houses the shutdown cooling system and ESF systems, such as the safety injection system (SIS), emergency feedwater system (EFWS), and the containment spray system (CSS).

The plant has an offsite and onsite power supply which are available to supply the unit auxiliaries during normal operations and the RPS and ESFAS during abnormal and accident conditions. The onsite power supply consists of the main generator, and generator circuit breaker, main unit transformer, batteries and two emergency diesel generators (EDGs). A non-safety grade combustion turbine generator (CTG) is available to supply power to safety-related components in the event both onsite and offsite ac power is lost.

Four high-head safety-injection pumps send borated water from the in-containment refueling water storage tank to mitigate accidents. Each of these provides sufficient borated water to direct vessel injection nozzles to acceptably limit the consequences of design basis accidents, such as loss-of-coolant accidents and main steamline breaks.

The steam and power conversion system converts the heat energy generated in the reactor into electrical energy. This system utilizes the main steam system, the high-pressure and low-pressure turbines, main generator, and a condensing cycle with regenerative feedwater heating. A turbine bypass system and atmospheric dump valves are available to dissipate heat from the reactor during a turbine trip, a reactor trip or both. A four-train, two-division EFWS provides emergency feedwater to the SGs in the event of loss of main feedwater.

The System 80+ design includes the Nuplex 80+ advanced control complex (ACC). The ACC design consists of the following five major interdependent systems:

- (1) Main control room panels — where plant is controlled.
- (2) Remote shutdown panel — includes two isolated redundant channels of safety-related instrumentation and controls necessary to achieve hot standby if the main control room must be evacuated.
- (3) Alarm and display system — displays and alarms to aid the operator in handling any challenges to important plant operations or safety functions.
- (4) Data processing system — a redundant computer based system which provides plant data and status information to the operational staff.
- (5) Component control system — controls discrete system components such as pumps, valves, heaters, and fans throughout the plant.

1.3 Comparison With Similar Facility Designs

The System 80+ standard design evolved from ABB-CE's System 80 design. The staff evaluated the System 80 design in NUREG-0852, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80," dated November 1981, and in supplements thereto. These reports are available for public inspection at the NRC Public Document Room at 2120 L Street, N.W., Washington, D.C. 20037.

Although the System 80+ standard design contains most of the features of the System 80 design, various engineering and operational improvements were included to provide additional safety margins and to address the Commission's severe accident, safety goal, and standardization policy statements. Some of the design differences between the System 80+ and the System 80 are listed below. Additional comparisons with the principal design features of the System 80+ standard design are provided in Section 1.3 of the CESSAR-DC.

System 80+ Reactor Design

- (1) an increased core thermal margin
- (2) a reduced hot leg temperature (T-hot)
- (3) part-strength CEAs for maneuvering without the use of soluble boron
- (4) extended CEA life
- (5) a ring-forged vessel
- (6) a lower end-of-life adjusted reference temperature

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System 80+ Pressurizer Design

- (1) an increased volume relative to power
- (2) surge line routing to minimize thermal stratification

System 80+ Steam Generator Design

- (1) an increased tube plugging margin
- (2) an increased boil dry time by increasing the secondary inventory
- (3) corrosion resistant tubes
- (4) higher steam quality
- (5) increased manway size

System 80+ Engineered Safety Systems Improvements

- (1) increased SIS redundancy — four trains separated into quadrants
- (2) increased ESF capacity relative to power
- (3) an in-containment refueling water storage tank, which eliminates automatic suction realignment and allows for reactor cavity flooding
- (4) safety depressurization system
- (5) eliminated automatic injection by the residual heat removal (RHR) pumps
- (6) eliminated chemical volume and control system (CVCS) safety functions
- (7) higher pressure rating of RHR
- (8) separate emergency and startup feedwater
- (9) eliminate automatic isolation of EFWS

System 80+ Containment Improvements

- (1) larger margin to ultimate strength
- (2) larger operating floor space
- (3) larger free volume
- (4) dual containment (pressure boundary plus a shield building with an annulus between)
- (5) equipment hatch sized for SG replacement
- (6) reactor cavity designed for severe accident mitigation

Instrumentation and Control System Improvements

- (1) advanced control, alarm, and display systems
- (2) control room design based on modern technology

1.4 Identification of Agents and Contractors

ABB-CE is the principal designer of the NSSS. Other organizations provided subcontracting services to ABB-CE as follows: Duke Engineering Services, Incorporated (DESI) is the principal designer of the nuclear annex structures, containment, and auxiliary and support systems. Stone and Webster Engineering Corporation (SWEC) is the principal designer for the steam and power conversion system, radwaste systems, and structures other than the nuclear annex. VECTRA Technologies, Inc. (formerly ABB-Impell) provided seismology and structural analysis support.

1.5 Summary of Principal Review Matters

The procedure for certifying a design is conducted in accordance with the requirements of subpart B of 10 CFR Part 52 and carried out in two stages. The technical review stage is initiated by an application filed in accordance with the requirements of 10 CFR 52.45, continues with reviews by the NRC staff and the ACRS, and concludes with the issuance of a final design approval (FDA) for certification. The administrative review stage begins with the publication of a Federal Register notice that initiates rulemaking, in accordance with Section 52.51, and records a proposed design certification rule. A draft proposed design certification rule was published in an advance notice of proposed rulemaking in the Federal Register (58 FR 58664) on November 3, 1993. The proposed rule will be issued after the FDA for design certification is issued. The rulemaking will be conducted by the Commission and also, in the event that there is a request for a hearing and the request is granted, by an Atomic Safety and Licensing Board. The rulemaking culminates with the denial or issuance of a design certification rule.

The staff performed its technical review of ABB-CE's application for certification of the System 80+ standard design in accordance with Commission guidance and the requirements of 10 CFR Sections 52.47 and 52.48. This FSER describes the results of the staff's technical review. The staff evaluated the technical information required by Section 52.47(a)(1)(i) in accordance with the standard review plan (NUREG-0800); that evaluation is the subject of this report. The staff's evaluation of the technically

relevant unresolved safety issues (USIs), generic safety issues (GSIs), and Three Mile Island requirements (Sections 52.47(a)(1)(ii) and (iv)) is provided in Chapter 20 of this report. The evaluation of the site parameters required by Section 52.47(a)(1)(iii) is provided in Chapter 2, and they are also discussed in Section 14.3. The staff's evaluation of the design-specific probabilistic risk assessment (Section 52.47(a)(1)(v)) is provided in Chapter 19. The evaluation of the inspections, tests, analyses, and acceptance criteria (ITAAC) required by Section 52.47(a)(1)(vi) is described in Section 14.3 of this report.

Interface requirements and representative conceptual designs (52.47(a)(1)(vii) through (ix)) are evaluated throughout Chapters 8 and 9 of this report, and are also discussed in Section 14.3. The staff also implemented the Commission's Severe Accident Policy Statement, dated August 8, 1985, and the Commission's SRMs on SECY-90-016 and SECY-93-087, in its resolution of severe accident issues. The staff's evaluation of severe accident issues is provided in Section 19.2 of this report.

The regulations in 10 CFR 52.47(a)(2) describe the level of design information needed to certify a standard design. Determining the acceptable level of design detail necessary for the staff to make its safety findings was one of the most challenging aspects of the staff's review. The SRM for SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," set forth the Commission's position on the level of design information that is required for a certification application, and the staff has followed that guidance in preparing this document. The staff has also followed the guidance of SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews." To allow for technology improvements and as-procured equipment characteristics, the staff based its safety determinations on the use of design acceptance criteria (DAC) for pipe stress analysis, radiation shielding and airborne concentrations, instrumentation

and controls (I&Cs), and control room design. The DAC are part of the CDM proposed for the System 80+ design. The staff's evaluation of the proposed CDM, including the DAC, is provided in Section 14.3 of this report.

As part of its technical review, the staff made numerous RAIs to gain sufficient bases for its safety findings, thereby meeting the requirement in Section 52.47(a)(3) to advise ABB-CE on the staff's requirements for additional technical information. See Appendix B for an index of ABB-CE's responses to these RAIs. The staff's evaluation of the scope of the design to be certified (Section 52.47(b)(1)) is provided in Section 1.2 of this report. The requirements set forth in Sections 52.47(b)(2) and (3) do not apply to the System 80+ standard design, because the System 80+ is an evolutionary reactor design that does not require prototype testing, and also is not a modular design, and is designed as a single unit, i.e., no safety systems would be shared at a multi-unit site.

The staff used the safety standards set forth in Section 52.48 for its technical review of the System 80+ standard design. In addition to these safety standards, the staff also followed Commission guidance provided in the SRMs for all applicable Commission papers, including those identified in Table 1.2. As a result of this guidance, the staff proposed design-specific regulations and justified exemptions from existing regulations in order to complete the framework of safety standards. An index to these safety standards is provided in Section 1.6 of this report.

Subsequent to the completion of the staff's review of the CESSAR-DC and CDM, ABB-CE will prepare a design control document (DCD). The DCD will consist of the CDM and Tier 2 information as described in Section 14.3 of this report. Applications that reference the certified System 80+ design will be required to conform with the DCD. The DCD will be available for public inspection at the NRC's Public Document Room when the proposed rule for design certification is published in the Federal Register.

Table 1.2 Commission papers applicable to CESSAR-DC

SECY No.	Title	Date
SECY-86-228	Introduction of Realistic Source Term Estimates Into Licensing	August 6, 1986
SECY-88-147	Integration Plan for Closure of Severe Accident Issues	May 25, 1988

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Table 1.2 Commission papers applicable to CESSAR-DC (continued)

SECY No.	Title	Date
SECY-89-012	Staff Plans for Accident Management Regulatory and Research Programs	January 18, 1989
SECY-89-013	Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)	January 19, 1989
SECY-89-230	Unresolved Safety Issue A-17, 'Systems Interactions in Nuclear Power Plants,'	August 1, 1989
SECY-89-341	Updated Light Water Reactor Source Term Methodology and Potential Regulatory Applications	Novmeber 6, 1989
SECY-90-016	Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements	January 12, 1990
SECY-90-065	Evolutionary and Passive Advanced Light Water Reactor Resources and Schedules	March 7, 1990
SECY-90-241	Level of Detail Required for Design Certification Under Part 52	July 11, 1990
SECY-90-307	Impacts of Source Term Timing on NRC Regulatory Positions	August 30, 1990
SECY-90-313	Status of Accident Management Programs and Plans for Implementation	September 5, 1990
SECY-90-341	Staff Study on Source Term Update and Decoupling Siting from Design	October 4, 1990
SECY-90-353	Licensing Review Basis for the Combustion Engineering, Inc., System 80+ Evolutionary Light Water Reactor	October 12, 1990
SECY-90-377	Requirements for Design Certification Under 10 CFR Part 52	November 8, 1990
SECY-91-078	Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light Water Reactor (LWR) Certification Issues	March 25, 1991

Table 1.2 Commission papers applicable to CESSAR-DC (continued)

SECY No.	Title	Date
SECY-91-135	Conclusions of the Probabilistic Seismic Hazard Studies Conducted for Nuclear Power Plants in the Eastern United States	May 14, 1991
SECY-91-161	Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions	May 31, 1991
SECY-91-178	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses	June 12, 1991
SECY-91-210	Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Requirements for Design Review and Issuance of a Final Design Approval	July 16, 1991
SECY-91-229	Severe Accident Mitigation Design Alternatives for Certified Standard Designs	July 31, 1991
SECY-91-262	Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs	August 16, 1991
SECY-91-272	Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants	August 27, 1991
SECY-91-292	Digital Computer Systems for Advanced Light Water Reactors	September 16, 1991
SECY-91-348	Issuance of Final Revision to Appendix J to 10 CFR Part 50, and Related Final Regulatory Guide 1.XXX (MS 021-5)	October 25, 1991
SECY-92-053	Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews	February 19, 1992
SECY-92-092	The Containment Performance Goal, External Events Sequences, and the Definition of Containment Failure for Advanced Light Water Reactors	March 17, 1992

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Table 1.2 Commission papers applicable to CESSAR-DC (continued)

SECY No.	Title	Date
SECY-92-127	Revised Accident Source Terms for Light Water Nuclear Power Plants	April 10, 1992
SECY-92-134	NRC Construction Inspection Program for Evolutionary and Advanced Reactors Under 10 CFR Part 52	April 15, 1992
SECY-92-170	Rulemaking Procedures for Design Certification	May 8, 1992
SECY-92-196	Development of Design Acceptance Criteria for the Advanced Boiling Water Reactor	May 28, 1992
SECY-92-214	Development of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications	June 11, 1992
SECY-92-287 and 287A	Form and Content for a Design Certification Rule	August 18, 1992 and March 26, 1993
SECY-92-292	Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future LWRs	August 21, 1992
SECY-92-299	Development of Design Acceptance Criteria for the Advanced Boiling Water Reactor in the Areas of Instrumentation and Controls and Control Room Design	August 27, 1992
SECY-92-327	Reviews of Inspections, Tests, and Analyses (ITAAC) for the GE Nuclear Energy Advanced Boiling Water Reactor (ABWR)	September 22, 1992
SECY-92-368	Final Rule Amending 10 CFR Part 52	October 29, 1992
SECY-92-381	Rulemaking Procedures for Design Certification	November 10, 1992
SECY-92-436	Status of Development of the NRC's New Construction Inspection	December 31, 1992

Table 1.2 Commission Papers Applicable to CESSAR-DC (Continued)

SECY No.	Title	Date
SECY-93-087	Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs	April 2, 1993
SECY-93-097	Integrated Review Schedules for the Evolutionary and Advanced Light-Water Reactor Projects	April 14, 1993
SECY-93-190	Regulatory Approach to Shutdown and Low-Power Operations	July 12, 1993

1.6 Index of Applicable Regulations and Exemptions

In accordance with 10 CFR 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing ABB-CE's application for design certification. During this review, the staff recognized that if certain regulations were applied to the System 80+ design, the use of those regulations would not serve the underlying purpose of the rule or would not be needed for achieving the fundamental intent of the rule. These exemptions are discussed in the sections of this report listed below.

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of individual design certifications to resolve selected technical and severe accident issues as "applicable regulations" for the advanced boiling water reactor and System 80+ standard designs. Among these issues are staff positions that deviate from or are not embodied in current regulations applicable to the System 80+ design. These issues were proposed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced LWR designs." The "applicable regulations" that are specific to the System 80+ design are identified and evaluated in this report. The proposed design certification rule will then include these "applicable regulations" for the System 80+ design for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63. These

"applicable regulations" are discussed in the sections of this report listed below.

<u>Section</u>	<u>Description of Exemption</u>
3.1.1	Exemption from operating-basis earthquake design requirement.
3.11.3.1	Exemption from 50.49(b)(3) changing RG 1.97, Revision 2 to Revision 3.
6.2.6	Exemptions from 10 CFR Part 50, Appendix J, assumptions for leak rate testing.
9.3.2	Exemption from post-accident sampling.
15A	Exemption from "TID 14844" Radiological Source Term.
18.6.1.3.4	Exemption from safety parameter display console requirement of 50.34(f)(2)(iv).
19.2.3.3.2.1.6	Exemption from dedicated containment penetration requirement.
<u>Section</u>	<u>Description of Applicable Regulation</u>
3.9.3.1.1, 6.3.2, 20 (GSI 105)	Applicable regulation for intersystem loss-of-coolant accident.

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<u>Section</u>	<u>Description of Applicable Regulation</u>
3.9.6, 6.5/6.6	Applicable regulation for inservice testing of pumps and valves.
7.1.3	Applicable regulation for digital instrumentation and control systems.
8.3.1	Applicable regulation for electric power system.
8.5, 20((USI A-44) blackout.	Applicable regulation for station
9.5.1	Applicable regulation for fire protection.
15.3.9 19.2.3.3.5.1	Applicable regulation for containment bypass potential resulting from steam generator tube ruptures.
17.3	Applicable regulation for a design reliability assurance program.
19.1.1.1	Applicable regulation for analysis of external events.
19.2.3.3.2.1	Applicable regulation for core debris cooling.
19.2.3.3.3.1	Applicable regulation for high-pressure melt ejection.
19.2.3.3.6	Applicable regulation for equipment survivability.
19.2.4	Applicable regulation for containment performance.
19.3.1	Applicable regulation for shutdown risk.

1.7 Interface Requirements

Section 52.47(1)(1)(vii) requires interface requirements that must be met by the site-specific elements of the non-certified portion of the plant design, such as the ultimate heat sink. The scope of the System 80+ design is discussed in CESSAR-DC Section 1.2 and is illustrated in CESSAR-DC Table 1.2-1. A discussion of the certified and non-certified portions of the System 80+ design are described in Section 1.2 of this report. CESSAR-DC Table 1.9 provides an index of the interface requirements for the System 80+ design.

1.8 Combined License Action Items

COL applicants and licensees who reference the certified System 80+ standard design in the future will be required to satisfy the requirements and commitments in the DCD. Also, certain requirements and commitments are identified in the CESSAR-DC as "COL License Information," and in this report as "COL Action Items." These COL action items relate to programs, procedures, and issues that are outside of the scope of the certified design review. An applicant for a COL will address each of these items in its application.

ABB-CE included the list of COL action items in Chapter 1 of the CESSAR-DC and provided an explanation of the items in the applicable sections of the CESSAR-DC. In the DSER, the staff had identified a number of COL action items from its review that ABB-CE has not identified. ABB-CE incorporated the COL action items that had been identified by the staff in the DSER and referred to these items in Table 1.10-1 of the CESSAR-DC as "COL license information." Confirmatory Item 1.10-1 is, therefore, resolved. Neither the table listings nor the descriptions within the cited CESSAR-DC sections are intended to constitute requirements for the COL applicant. The requirements are provided in the appropriate governing regulations.

1.9 Index of Tier 2* Information

The staff has determined that the CESSAR-DC commitments that are discussed in the following sections of this report, if considered for a change by COL applicant or licensee that references the certified System 80+ design, will require NRC review and approval prior to implementation of the change. The commitments identified below will be listed in the proposed rule certifying the System 80+ design as Tier 2* information.

<u>CESSAR-DC Section</u>	<u>FSER Section</u>	<u>Description</u>
3.8.2	3.8.2	ASME Code Section III, Subsection NE, (1989 Edition) for containment design and construction
3.8.3, 3.8.4 3.8.5	3.8.3, 3.8.4	ACI-349 (1985 Edition) for design and construction of internal structures
3.8.3, 3.8.4	3.8.3, 3.8.4	N690 (1984 Edition) for structural design and construction

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<u>CESSAR-DC Section</u>	<u>FSER Section</u>	<u>Description</u>	<u>CESSAR-DC Section</u>	<u>FSER Section</u>	<u>Description</u>
3.9.6	3.9.6.2.1	Design, qualification, and preoperational testing for motor-operated valves	18.7	18.9	Information presentation and panel layout evaluation
3.10	3.10.1	Equipment seismic qualification methods	18.8	18.9	Control and monitoring outside the main control room
N.A.	3.12	Piping DAC	18.9	18.9	Verification and validation
4.2	4.2	First cycle fuel, control rod and design and the methods used to analyze these components			
7.2.2.3.2	7.1.2	Identification of review criteria (setpoint methodology)			
7.1.2.32	7.1.4.6, 7.1.4.8	Hardware and software changes, software program manual, software development process			
7.1.2.32, 7.3.1.1.6	7.1.4.9	Commercial dedication			
7.1.2.5, 7.2.1.2 7.3.1.2	7.2.2	Equipment qualification			
7.2.1.2, 7.3.1.2	7.1.2	Electromagnetic compatibility			
7.3.1.2	7.3.10	ESFAS Hardware and software changes			
7.4.1.1.10	7.4.6	Emergency shutdown from outside the control room, (master transfer switch)			
18.5	18.9	Functional task analysis			
18.6	18.9	Control room configuration			

2 SITE CHARACTERISTICS

Introduction

The staff reviewed the site parameters required by 10 CFR 52.47 (a)(1)(iii) and contained in CESSAR-DC Chapter 2, including the envelope of plant site parameters for the System 80+ design in plant CESSAR-DC Table 2.0-1. The staff found that the list of site parameters is consistent with that which is contained in appropriate sections of the Standard Review Plan (SRP) Chapter 2 and Title 10 of the Code of Federal Regulations, 10 CFR Parts 50, 52, and 100. In its review of the System 80+ standard design, the staff finds that the acceptance criteria provided in the SRP for the review of site suitability for the site parameters shown in CESSAR-DC Table 2.0-1, "Envelope of Plant Site Design Parameters," are sufficient.

The staff has not identified any unique features of the System 80+ design that would require additional limitations for the selection of sites compatible with the System 80+ design. Therefore, the combined license (COL) applicant referencing the certified System 80+ design may use the applicable SRP guidelines to evaluate the suitability of its site for the construction of the System 80+ certified design. To ensure that the System 80+ design envelopes by the site-related parameters, the staff will perform a detailed review of site parameters during the COL application phase. The site-specific information to be provided by the COL applicant referencing the System 80+ design is discussed below.

In its letter dated March 27, 1992, ABB-CE stated that the site-specific safety analysis reports (SARs), submitted by the COL applicant, will demonstrate that the parameters specified in CESSAR-DC Table 2.0-1 are met. This was DSER Confirmatory Item 2.4.3-1 and was also discussed in Sections 2.4.4 to 2.4.11 of the DSER. The letter from ABB-CE resolved this DSER Confirmatory item. This action by the COL applicant is COL Action Item 2.0-1.

It should be noted that the site-specific parameters, which are discussed here as required at the time of a plant-specific COL application, may also be submitted and considered by the NRC in connection with an application for an Early Site Permit under 10 CFR Part 52, Subpart A. The standard plant design certification is Subpart B of 10 CFR Part 52.

2.1 Geography and Demography

The COL applicant shall submit specific information related to site location and description, exclusion area authority and control, and population distribution as discussed in this section and listed in Table 1.10-1 of CESSAR-DC. ABB-CE discussed this in CESSAR-DC

Section 2.1. This is acceptable. This is COL Action Item 2.1-1.

2.1.1 Site Location and Description

The COL applicant shall submit site-specific information related to site location, including political subdivisions, natural and man-made features, population, highways, railways, waterways, and other significant features of the area. ABB-CE included this in Section 2.1.1 of CESSAR-DC.

2.1.2 Exclusion Area Authority and Control

The COL applicant shall provide site-specific information related to activities that may be permitted within the designated exclusion area. ABB-CE included this in Section 2.1.2 of CESSAR-DC.

2.1.3 Population Distribution

The COL applicant shall provide population distribution data for the site environs. ABB-CE included this in Section 2.1.3 of CESSAR-DC.

2.2 Nearby Industrial, Transportation, and Military Facilities

Site-specific industrial, transportation, and military hazards are discussed in Section 2.2 of CESSAR-DC. The COL applicant shall submit site-specific information on nearby industrial, transportation and military facilities as discussed in this section. This is acceptable. This is COL Action Item 2.2-1.

2.2.1 Aircraft Hazards

As stated in Section 2.2.1 of CESSAR-DC, ABB-CE proposes that a site for the System 80+ plant be accepted without further review if the distances from the plant meet the following requirements which are from SRP Section 3.5.1.6, "Aircraft hazards":

- (1) The plant-to-airport distance (D) is between 8 and 16 kilometers (5 and 10 miles), and the projected annual number of operations is less than $195 D^2$ ($500 D^2$ for a distance between 5 and 10 miles); or if D is greater than 16 kilometers (10 miles), and the projected annual number of operations is less than $390 D^2$ ($1000 D^2$ for a distance greater than 10 miles).
- (2) The plant is at least 8 kilometers (5 miles) from the edge of military training routes including low-level training routes, except for those associated with a

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usage greater than 1000 flights per year, or where activities (such as practice bombing) may create unusual stress conditions.

- (3) The plant is at least 3.2 kilometers (2 miles) beyond the nearest edge of a Federal airway, holding pattern, or airport.

If these site proximity acceptance criteria are not met, or if sufficiently hazardous military activities are identified, the COL applicant will be required to submit a detailed review of the aircraft hazards in the COL application. These parameters are included in CESSAR-DC Table 2.0-1.

The proposed site requirements in CESSAR-DC Section 2.1.1 meet the criteria of SRP Section 3.5.1.6 and are, therefore, acceptable. The criteria of SRP Section 3.5.1.6 are based on the assumption that adherence to these criteria results in very low offsite release probability (less than 10^{-7} /reactor-year).

2.2.2 Transportation

The COL applicant will be required to submit site-specific information on nearby transportation. ABB-CE included this in Section 2.2.2 of CESSAR-DC.

2.2.3 Other Industrial Hazards

The COL applicant will be required to submit specific information on potential industrial hazards on and offsite. ABB-CE included this in Section 2.2.3 of CESSAR-DC.

2.3 Meteorology

ABB-CE specifies in Section 2.3 of CESSAR-DC bounding conditions and assumptions related to regional climatology, local meteorology, onsite meteorological measurements program, short-term (accident) diffusion estimates χ/Q , and long-term (routine) diffusion estimates χ/Q and onsite (accident) diffusion estimates χ/Q . The term χ/Q is the relative atmospheric concentration of radiological releases at the receptor point in terms of the rate of release from the point of release.

ABB-CE states that site-specific meteorology information would be provided by the COL applicant. Further, ABB-CE noted that, if the site-specific meteorology parameters are outside the values in CESSAR-DC Table 2.0-1, any alternatives required for siting the System 80+ design will be presented in the site-specific SARs. This is in Section 2.3 of CESSAR-DC. This is acceptable. This is COL Action Item 2.3-1.

2.3.1 Regional Climatology

The COL applicant shall submit site-specific information related to CESSAR-DC Section 2.3.1, "Regional Climatology."

2.3.2 Local Meteorology

The COL applicant shall submit site-specific information related to CESSAR-DC Section 2.3.2, "Local Meteorology."

2.3.2.1 Tornadoes

ABB-CE specified in Table 2.0-1 of CESSAR-DC the following criteria regarding tornadoes (i.e., the site tornado design basis):

Maximum Tornado Wind Speed	330 mph	531 km/h
Translational Velocity	70 mph	113 km/h
Radius	150 ft	45.7m
Maximum Pressure Differential	2.4 psi	16.5 kPa
Missile Spectra	per SRP 3.5.1.4 Spectrum II	
Rate of Pressure Drop	1.7 psi/sec	11.7 kPa/sec
Rotational Speed	260 mph	418 km/h

The current NRC regulatory position with regard to a design-basis tornado is in WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," May 1974, and Regulatory Guide (RG) 1.76, "Design Basis Tornado for Nuclear Power Reactors." WASH-1300 stated that the probability of occurrence of a tornado that exceeds the design basis tornado (DBT) should be on the order of 10^{-7} per year per nuclear power plant. The RG delineates the maximum wind speeds of 140 to 360 mph depending on the region of the contiguous 48 States.

The staff reevaluated the regulatory positions in RG 1.76 using tornado data which was not available when the RG was developed. The staff's evaluation was published as NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," dated May 1986, and included the tornado data tape prepared by the National Severe Storm Forecast Center with 30 years of data, 1954 through 1983. This tape contains the data for the approximately 30,000 tornadoes that occurred during the period.

The reevaluation found that the tornado strike probabilities range from near

10^{-7} per year for much of the western United States to about 10^{-3} per year in the central United States. Thus, windspeed values associated with a tornado having an annual strike probability of 10^{-7} range from less than 153 mph to 332 mph (less than 246 km/h to 534 km/h). These windspeed estimates are 50 to 100 mph (80 to 160 km/h)

lower than the speed estimates presented in WASH-1300 and RG 1.76 for most of the United States. On the basis of this analysis, NUREG/CR-4461 concluded that it is reasonable to reduce tornado design-basis windspeed to 200 mph (322 km/h) for the United States west of the Rocky Mountains and to 330 mph (531 km/h) east of the Rocky Mountains.

Therefore, the staff concludes that the tornado design basis specified above by ABB-CE for the System 80+ design is a conservative boundary for a tornado having an annual strike probability of 10^{-7} and is acceptable. The staff further finds that the translational speed, the radius of maximum rotational speed, the rate of pressure drop, the rotational speed, and the maximum pressure drop specified above is also acceptable.

It should be recognized, however, the tornado design-basis requirements have been used in establishing structural requirements (minimum concrete wall thickness) for the protection of nuclear plant safety-related structures, systems, and components (SSCs) against the effects not covered explicitly in review guidance such as RGs or the SRP. Specifically, some aviation (general aviation light aircraft) crashes, nearby explosions, and explosion debris or missiles have been reviewed and evaluated routinely by the staff by considering the existence of the tornado protection requirements. Hence, the staff's acceptance of a COL application will also necessitate a concurrent review and evaluation of the effect on the protection criteria for some external impact hazards, such as general aviation or nearby explosions.

2.3.3 Onsite Meteorological Measurements Program

Details on the atmospheric diffusion characteristics of a nuclear power plant site are required to determine that postulated accidental, as well as routine operational, releases of radioactive materials are within NRC regulatory guidelines. The meteorological characteristics of a site are determined by staff evaluation of meteorological data collected at the site by the onsite meteorological measurements program in accordance with RG 1.23, "Onsite Meteorological Programs."

The onsite meteorological measurements program is discussed in CESSAR-DC Section 2.3.3. The COL applicant will submit the onsite meteorological measurements program for review by the staff. This is acceptable. This is included in COL Action Item 2.3-1.

2.3.4 Short-Term (Accident) Diffusion Estimates (χ/Q)

In lieu of site specific meteorological data, ABB-CE provided a bounding set of short-term (accident) atmo-

spheric relative concentrations χ/Q s for the System 80+ design. The meteorological data representative of an 80-90th percentile of U.S. operating nuclear power plant sites were used to develop these χ/Q values. ABB-CE calculated ground-level 0-2 hour χ/Q values at 0.5 mile Exclusion Area Boundary (EAB) using Gaussian diffusion model modified for source configuration and lateral plume meander under neutral and/or stable atmospheric conditions. In calculating these χ/Q values, ABB-CE used the methodology provided in RG 1.145, "Atmospheric Dispersion Model for Potential Accident Consequence Assessment at Nuclear Power Plants."

For time periods greater than 2 hours (i.e., 0-8 hours, 8-24 hours, 1-4 days, and 4-30 days), ABB-CE determined χ/Q values at a 2.0 mile low population zone (LPZ) using logarithmic interpolation method given in RG 1.145. Table 2.3-1 of CESSAR-DC provides these short-term (accident) χ/Q values at both the EAB and LPZ receptors, as follows:

Location	Time Period	Dilution Factor χ/Q (sec/m^3)
EAB	0-2 hours	1.00×10^{-3}
LPZ	0-8 hours	1.35×10^{-4}
LPZ	8-24 hours	1.00×10^{-4}
LPZ	1-4 days	5.40×10^{-5}
LPZ	4-30 days	2.20×10^{-5}

A site selected for use of a System 80+ facility should have χ/Q values within the bounds specified above. In the event that a site selected for the System 80+ design exceeds the bounding χ/Q values, the COL applicant must demonstrate that the radiological consequences associated with the controlling design basis accident using its site-specific χ/Q values continues to meet the dose reference values given in 10 CFR Part 100 and control room operator dose limits given in GDC 19.

2.3.5 Long-Term (Routine) Diffusion Estimates (χ/Q)

ABB-CE also determined long-term relative concentrations at the LPZ boundary. An annual average χ/Q value at the LPZ was calculated using the method described in RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1 (July 1977).

The staff finds that ABB-CE's method for determining long-term (routine) values for χ/Q were calculated using the methodology given in RG 1.111; therefore, these values represent the values of atmospheric dispersion at the

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LPZ expected to occur at a selected site and are acceptable.

2.3.6 Onsite (Accident) Diffusion Estimates (χ/Q).

In Section 2.3.6 of CESSAR-DC, ABB-CE calculated the on-site accident one hour χ/Q values using a time-based building wake model. This model was developed by Ramsdell (Ramsdell, J.V., "Diffusion in Building Wakes for Ground-Level Releases," Atmospheric Environment, Volume 24B, No. 3, pp 377-388, 1990). Solely for the purposes of determining the χ/Q value associated with the limiting accident scenario, ABB-CE used a meteorological database that was characterized as "representative of an 80-90th percentile of U.S. commercial nuclear power plant sites." The resulting χ/Q values were provided in CESSAR-DC Tables 2.3-2 through 2.3-5.

To perform an independent assessment of the on-site (accident) diffusion estimates (χ/Q s) for the control room habitability assessment, Pacific Northwest Laboratories (PNL) conducted a peer review meeting for NRC to assess the 1990 Ramsdell Model. At that meeting, the peer group expressed concerns regarding the model's treatment of receptors within three building heights of the release point. The peer review group identified information for close-in distances that could be used to evaluate the performance of the 1990 Ramsdell Model. This work is the near-source dilution model developed by Wilson and Chui (Wilson, D.J. and E.H. Chui, Influence of Building Size on Rooftop Dispersion of Exhaust Gas, Atmospheric Environment, Volume 28, 1994, accepted for publication).

One objective of the Wilson-Chui Model is to predict a lower-bound minimum dilution function (i.e., maximum atmospheric relative concentrations). The field measurements from the experimental data used to derive the 1990 Ramsdell Model were used to evaluate the performance of the Wilson-Chui Model as a predictive tool. The results of this comparison showed that use of the Wilson-Chui Model resulted in χ/Q values which were greater than the experimental data in every case. For the purposes of evaluating ABB-CE's use of the 1990 Ramsdell Model, the staff considered the Wilson-Chui Model results at a performance level whereby it would not be exceeded by more than 5% of the experimental data.

In conjunction with the 95th percentile Wilson-Chui Model χ/Q values, the staff used dose conversion factors based on International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers." The staff determined that, using the Wilson-Chui Model/ICRP Publication 30 framework, the control room habitability aspects of the System 80+ design are acceptable. Control room habitability

is discussed in Section 6.4 of this report. For the System 80+ design, as long as the linkage exists between the use of the 1990 Ramsdell Model and the dose conversion factors based on ICRP Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation," the staff concludes that the resulting on-site (accident) diffusion estimates (χ/Q s) provided in CESSAR-DC Tables 2.3-2 through 2.3-5 are acceptable.

2.4 Hydrological Engineering

Hydrological engineering parameters for a site are discussed in Section 2.4 of CESSAR-DC. The COL applicant shall submit site-specific information on hydrologic engineering as discussed in this section and listed in Table 1.10-1 of CESSAR-DC. This is acceptable. This COL Action Item 2.4-1.

ABB-CE was requested to revise Enclosure 2 of its letter dated March 27, 1992, to remove the reference to certain hydraulic engineering subjects not addressed in the CESSAR-DC or add evaluations of these subjects from Enclosure 2 to the CESSAR-DC. This was DSER Open Item 2.4.14-1. ABB-CE has adequately added information to Section 2.4 of CESSAR-DC on these subjects. This resolved DSER Open Item 2.4.14-1.

2.4.1 External Floods

The COL applicant will be required to submit site-specific information on external floods. ABB-CE included this in Section 2.4 of CESSAR-DC. This information is necessary to establish the suitability of the proposed site for the certified design by showing that the site-specific external flood level is below that used for the certified design.

2.4.2 Internal Floods

This topic is discussed in Section 3.4, "Water Level (Flood) Design," of this report.

2.4.3 Probable Maximum Flood on Streams and Rivers

The COL applicant will be required to submit site-specific information on the probable maximum flood.

2.4.4 Potential Dam Failures

The COL applicant will be required to submit site-specific information on potential dam failures.

2.4.5 Probable Maximum Surge and Seiche Flooding

The COL applicant will be required to submit site-specific information on probable maximum surge and seiche flooding.

2.4.6 Probable Maximum Tsunami Loading

The COL applicant will be required to submit site-specific information on probable maximum tsunami loading.

2.4.7 Ice Effects

The COL applicant will be required to submit site-specific information on effects of ice.

2.4.8 Cooling Water Canals and Reservoirs

The COL applicant will be required to submit site-specific information on cooling water canals and reservoirs.

2.4.9 Channel Diversions

The COL applicant will be required to submit site-specific information on channel diversions.

2.4.10 Flood Protection Requirements

The COL applicant will be required to submit site-specific information on flood protection requirements.

2.4.11 Cooling Water Supply

The COL applicant will be required to submit site-specific information on the cooling water supply. Cooling water inlet temperatures are specified in CESSAR-DC Table 2.0-1.

2.4.12 Groundwater

The COL applicant will be required to provide site-specific information on groundwater. CESSAR-DC Table 2.0-1 specifies the maximum groundwater level to be 0.6 m (2 ft) below grade. The staff reviewed this site parameter in Section 2.6 of this report and found it acceptable.

2.4.13 Accidental Release of Liquid Effluents in Ground and Surface Water

The COL applicant will be required to submit site-specific information on accidental release of liquid effluents in ground and surface water.

2.4.14 Technical Specifications and Emergency Operation Requirements

The COL applicant will be required to provide site-specific information on technical specification and emergency operation requirements on hydrologic engineering.

2.5 Geology, Seismology, and Geotechnical Engineering

The COL applicant will submit site-specific information related to basic geologic and seismic information, vibratory ground motion, surface faulting, and stability of subsurface materials and foundations as discussed in the following sections and Table 1.10-1 of CESSAR-DC. This is acceptable. This is COL Action Item 2.5-1.

2.5.1 Basic Geological and Seismic Information

The COL applicant will submit site-specific information related to regional and site physiography, geomorphology, stratigraphy, lithology, and tectonics. ABB-CE included this in Section 2.5.1 of CESSAR-DC.

2.5.2 Vibratory Ground Motion

The COL applicant will submit site-specific information related to seismic, geologic, and tectonic parameters of the site and region, correlation of earthquake activity with geologic structure or tectonic provinces, maximum earthquake potential, seismic wave transmission characteristics of the site, and the safe shutdown earthquake (SSE). ABB-CE included this in Section 2.5.2 of CESSAR-DC.

2.5.2.1 Seismicity

The complete historical record of earthquakes in the region will be included in the site-specific data. The COL applicant will submit all available information pertaining to and concerning epicenter coordinates, depth of focus, origin time, highest intensity, magnitude, seismic moment, source mechanism, source dimensions, distance from site, strong motion recordings, and earthquake-induced geologic failures for each event.

2.5.2.2 Geologic and Tectonic Characteristics of Site and Region

The COL applicant will identify tectonic provinces based on the development and characteristics of the current tectonic regime of the region and the pattern and level of historic seismicity.

Site Characteristics

2.5.2.3 Correlation of Earthquake Activity with Geologic Structure or Tectonic Provinces

The COL applicant will determine the relationship between earthquake activity history and the geologic structure or tectonic provinces of the region.

2.5.2.4 Maximum Earthquake Potential

The COL applicant will show that the free field control motion described in CESSAR-DC Section 2.5.2.5 envelops the maximum possible vibratory ground motion at the site, by the site-specific analysis. This is discussed in Section 2.5.2.6 of this report.

2.5.2.5 Seismic Wave Transmission Characteristics of the Site

2.5.2.5.1 Control Motions

The ABB-CE System 80+ SSE design includes the application of three control motion spectral shapes (CMS1, CMS2 and CMS3) all anchored at 0.3-g horizontal peak ground acceleration (Figure 2.1 of this report). The three spectral shapes are the following:

1. Control motion CMS1 is the design spectrum from RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants." CMS1 was used to cover both deep soil and rock sites. It is applied at the free ground surface. Because CMS1 is a standardized response spectrum shape, ABB-CE states in CESSAR-DC that it is considered an appropriate control motion for both rock and soil sites.
2. Control motion CMS2 was used to cover shallow and intermediate depth soil sites as well as rock sites. It is applied at a hypothetical rock outcrop. The CMS2 spectrum is lower than the CMS1 and CMS3 spectra at frequencies below 3 Hz. The CMS2 spectrum is higher than CMS1 and CMS3 at frequencies above about 10 Hz to account for high-frequency ground motions observed for earthquakes in the eastern United States (EUS). The time-history associated with this motion was used as a rock outcrop motion and convolved upward through generic soil columns to yield a broad range of expected responses (see discussion in Section 2.5.2.5.2 of this report).
3. Control motion CMS3 was also used to cover shallow and intermediate depth soil sites as well as rock sites. It is also applied at a hypothetical rock outcrop. CMS3 is based on the median spectra in NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Plants," dated May 31, 1978 to cover

lower frequency motions which may not be covered by CMS2. The high-frequency range is enhanced over the NUREG/CR-0098 spectrum up to a frequency of 15 Hz. The amplifications at these higher frequencies are not as great, however, as the CMS2 spectrum.

The COL applicant must demonstrate that the three control motions discussed above adequately envelope the site parameters. This is discussed in Section 2.5.2.6 of this report and is included in COL Action Item 2.5-1.

In evaluating the control motions, the staff has the following comments:

DSER Open Item 2.5.2.5.1-1 regarding acceptance criteria of SRP Section 3.7.1, "Seismic Design Parameters", for the 7-percent damping is addressed and closed out in Section 3.7.1 of this report.

ABB-CE responded to the DSER Open Item 2.5.2.5.1-2 regarding the problem of a significant "valley" occurring in the foundation spectra in its letter dated December 23, 1992. It is stated in that letter that the reduction of spectra at the free-field foundation level occurs at a frequency corresponding approximately to that of the sublayer between the ground surface (where the CMS1 motion is specified) and the foundation level. To overcome this reduction, the deconvolution is done considering a wide range of modulus values of the soil comprising this layer. This is acceptable and this DSER open item is closed.

ABB-CE discusses the above control motions, CMS1, CMS2, and CMS3, in CESSAR-DC Section 2.5.2.5.1 and indicates how they will be used to generate seismic input for the design of Category I structures. This is acceptable. Therefore, DSER Open Items 2.5.2.5.1-3 and 2.5.2.5.1-4 are resolved.

ABB-CE explains how the COL applicant is to use the three control motion spectra in conjunction with the site-specific spectra to determine the acceptability of the site in Section 2.5.2.5.1 of CESSAR-DC. This is acceptable. Therefore, DSER Open Item 2.5-1 is resolved.

ABB-CE states that the System 80+ seismic design will cover the majority of potential sites in the continental United States. The possible exceptions are sites near active faults which could result in free field spectra higher than the design basis. The staff notes that there are only two currently operating nuclear power plant sites in the United States that have SSE ground acceleration values higher than the envelope of the three control motion spectra used in the certification of the System 80+ design. A discussion of the application of the above three control

Figure 2.1 Control Motions for System 80+ Seismic Design (CMS1, CMS2, CMS3)

(Note: This is the same as Fig. 2.5-5 in CESSAR-DC.)

Site Characteristics

motion spectra to the plant seismic design criteria is in Section 3.7.2 of this report. The COL applicant must perform a site-specific soil column analysis to calculate response spectra at the ground surface and foundation level using the site-specific control motion and soil condition. This analysis is needed to verify the satisfaction of the following condition:

If the site-specific rock outcrop spectra indicate exceedances at some frequencies of interest, the computed site-specific free-field spectra at ground surface will be compared with the envelope of surface spectra from all cases analyzed for the generic site study in the CESSAR-DC. If the site-specific, free-field surface spectra are enveloped by these generic envelope spectra, the site is considered acceptable. This is included in COL Action Item 2.5-1.

2.5.2.5.2 Generic Soil Sites

A set of 13 sites was developed (12 soil sites and 1 rock outcrop site) with the aim of providing a broad range of potential site conditions for use in the generic system design. The resulting ground-motion estimates provide a broad range of responses over the frequency range of interest for the system design. However, the following additional aspects of the problem need to be considered by the COL applicant performing the site-specific soil column analysis:

- (1) The range in potential rock shear wave velocity used in a sensitivity study in response to a staff question (Q230.2 of Appendix B of this report) submitted on September 26, 1991, was limited to 2440 m/sec (8000 ft/sec). The staff noted in the meetings of April 27 through April 30, 1992, (NRC meeting summary dated June 3, 1992) with ABB-CE that rock velocities on the order of 3660 m/sec (12,000 ft/sec) had been measured at EUS sites founded on hard crystalline rocks. Should the site-specific rock velocities exceed 2440 m/sec (8000 ft/sec), the COL applicant should do an analysis using the actual rock velocity of the prospective site.
- (2) At deep-soil sites (on the order of 91 m (300 ft) deep or more), the effect of additional confinement produced by overburden leads to an effective stiffening of the soils at the deeper depths. Both of these effects can lead to higher peak responses at the ground surface, and lower strains in the soil columns than considered in the generic evaluations. Should the soil depth exceed 91 m (300 ft), the COL applicant should account for the effects of additional confinement and effective stiffening of

the soil at deeper depths when developing the surface motions to evaluate the adequacy of the standard design.

The COL applicant referencing the certified System 80+ design will be required to develop site-specific geological, seismological, and geotechnical data and will submit these data to the NRC staff for review. These data should be comparable to the design-basis assumptions regarding the SSE, including the verification of the ground-motion response spectra. The staff will compare the site-specific earthquake ground-motions to the ground motions used as input for the design certification in accordance with procedures and acceptance criteria in SRP Section 2.5.2, "Vibratory Ground Motion." ABB-CE included this in Sections 2.5, 2.5.1, 2.5.2, 2.5.2.1, and 2.5.2.1 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

For a shallow soil site, to confirm that the site-specific conditions are satisfied and the standard design is suitable for a specific site, the COL applicant may define the site-specific ground motion (ground response spectra) at ground surface in the free field, if this ground motion is developed from a statistical analysis of a sufficient number of ground-motion records that are chosen for their similarity in source path and site properties, as well as magnitude, fault type, and tectonic environment, and can be classified as a site-specific (shallow soil site) surface ground motion. In all cases, the appropriate level of the site-specific spectrum for comparison with the design certification spectrum is the 84th percentile of the statistical analysis of the earthquake data. However, if the shallow site surface motion cannot be developed in accordance with these criteria, the COL applicant should follow the steps given below to confirm the suitability of the proposed site for the certified plant:

- (1) Define a site-specific ground motion as the free-field motion at a level that complies with the guidelines of SRP Section 3.7.1, for example, at a rock outcrop or a hypothetical rock outcrop.
- (2) Compare the site-specific ground motion with the input motion specified on the outcrop for the standard plant design.
- (3) Determine if the site-specific input motion at the control point exceeds the standard design outcrop motion.
- (4) Perform site-specific soil column convolution analysis to develop the free field ground surface motions for comparison with the design basis surface spectra.

ABB-CE states this in Section 2.5 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.2.6 Safe Shutdown Earthquake

The COL applicant will compare site-specific earthquake ground motions to the ground motions used as input for the design certification. The COL applicant must verify that the site-specific design response spectra are enveloped by the control motion spectra CMS1, CMS2, or CMS3. ABB-CE included this in Section 2.5.2 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

For control motion spectra CMS2 and CMS3, the vertical spectrum is defined to be two-thirds of the corresponding horizontal spectrum over the entire frequency range of interest. This two-thirds factor may constrain the site selection by the COL applicant. However, both the horizontal and vertical free field surface design spectra must envelope the 84th percentile spectra that would be developed from any site-specific geological and seismological evaluation (SRP Section 2.5.2).

2.5.2.7 Operating Basis Earthquake

The operating basis earthquake (OBE) is not considered in the design basis. This approach is acceptable since the Commission has eliminated the OBE from the design for evolutionary reactors, per SECY-90-016, "Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990. See the discussion of this approach in Section 3.1 of this report.

2.5.2.8 Site Response

Section 2.5.2.7.1 of CESSAR-DC addresses the procedures used to perform convolution calculations for vertically propagating shear wave and compression waves. For the shear wave problem, the method of calculation used by ABB-CE has been a standard practice over the years to achieve what is reviewed as reasonable estimates of site response to the specified ground-motion estimates. Laboratory results are available for use in the selection of appropriate soil properties, and a significant number of comparisons have been made by a variety of investigators to correlate measured and computed responses. ABB-CE used the synthetic time-histories H1 of motion CMS2 as the input rock outcrop motion for each soil case, and obtained the strain-compatible modulus and damping values for that soil case. These properties were then used without

further modifications for the analysis involving synthetic time-history H2 of CMS2 and time-histories H1 and H2 of CMS1 and CMS3 as the input rock outcrop motions.

For the compressional wave problem, no significant laboratory data are available on which to base estimates of soil properties, and no significant effort has been made to compare measured and computed responses. ABB-CE has, therefore, performed the vertical (compressional wave) motion analyses by converting the strain-compatible shear moduli (from the shear wave analysis) to constrained moduli assuming a Poisson's ratio of 0.4 for all soil layers. The strain-compatible damping values from the shear wave analysis were multiplied by 1/3 to obtain an estimate of the damping associated with the compressional waves.

In response to a staff question regarding the effect of the variation of

Poisson's ratio on the vertical soil-structure interaction (SSI) analyses, ABB-CE performed a parametric study in which the seismic response at key locations such as the ground surface was determined using Poisson's ratio values of 0.3 and 0.47. The results of this study are discussed in Section 3.7.2 of this report. As stated therein, ABB-CE contends that, regardless of the damping and Poisson's ratio used in the certification application, the site can be considered as qualified as long as the vertical motion at the surface as developed by the COL applicant for site-specific conditions is enveloped by the System 80+ surface spectra included in the CESSAR-DC. Based on the results of this study, the staff concludes that the COL applicant must address the impact of the actual site-specific Poisson's ratio and damping values and their variability on the vertical seismic response. This is acceptable. This is included in COL Action Item 2.5-1. Therefore, DSER Open Item 2.5.2.8-1 is resolved.

2.5.3 Surface Faulting

The COL applicant shall submit detailed surface and subsurface geological and geophysical information to ensure that no potential exists for surface faulting affecting the site. ABB-CE included this in Section 2.5.3 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

The NRC stated in the DSER that it would not consider certifying the System 80+ design to withstand surface faulting. ABB-CE states in Section 2.5.3 of CESSAR-DC that the System 80+ plants will not be built on sites with surface faulting. This is acceptable. Therefore, DSER Open Item 2.5.3-1 is resolved.

Site Characteristics

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 Geologic Features

The COL applicant will submit site-specific information regarding the geologic features underlying the site. ABB-CE included this in Sections 2.5.4 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.4.2 Properties of Underlying Materials

The COL applicant will use state-of-the-art methods to determine the static and dynamic engineering properties of all foundation soils and rocks in the site area. ABB-CE included this in Section 2.5.4.2 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

The COL applicant must verify that the minimum shear wave velocity of foundation soils is not less than the values documented in the CESSAR-DC. ABB-CE addresses this item by stating in CESSAR-DC Table 2.0-1 that a site has to meet all three of the following criteria: minimum foundation level soil shear wave velocity of 213 m/sec (700 ft/sec), minimum foundation level soil bearing capacity of 59 ton/m² (12 ksf), and no soil liquefaction potential when subjected to the site-specific SSE ground motion. This is acceptable. This is included in COL Action Item 2.0-1, which is discussed at the end of Section 2.6 of this report.

2.5.4.3 Relationship of Foundation to Underlying Materials

The COL applicant must submit all data pertaining to site-specific soil layers (including their thicknesses, densities, moduli, and Poisson's ratios) between the basemat and the underlying rock stratum. Plot plans and profiles of site explorations will be provided in the site-specific SAR. ABB-CE included this in Section 2.5.4.3 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.4.4 Soil and Rock Characteristics

The COL applicant must perform geophysical and geotechnical investigations at the site, and submit the engineering classifications and descriptions of soils and rocks supporting the structural foundations. ABB-CE included this in Section 2.5.4.4 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.4.5 Excavation and Backfill

The COL applicant referencing the System 80+ certified design must submit data concerning the extent (horizontally and vertically) of all seismic Category I excavations, and backfills. The sources, quantities, and static and dynamic engineering properties of borrow materials will be described in the site-specific SAR. The compaction requirements, results of field compaction tests, and fill material properties such as moisture content, density, permeability, compressibility, and gradation should also be submitted. ABB-CE included this in Section 2.5.4.5 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.4.6 Groundwater Conditions

The COL applicant must submit the critical cases of groundwater conditions relative to the foundation stability of the safety-related structures at the site. They will also confirm that the soil properties of the various layers under all possible groundwater conditions during the plant life will fall within the range of values assumed in the CESSAR-DC. ABB-CE included this in Section 2.5.4.6 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.4.7 Response of Soil and Rock to Dynamic Loading

ABB-CE presents standard curves in CESSAR-DC Section 2.5 showing the variation of shear modulus and material damping with shear strain for soils based on the publications by H. B. Seed and I. M. Idriss, "Soil Moduli and Damping Factors for Dynamic Response Analysis," Report No. EERC 70-10, University of California, Earthquake Research Center, Berkeley, California, December 1970.

From these sets of curves, ABB-CE chose the upper curve for the shear modulus variation and the lower curve for the damping variation. In response to the staff question Q 230.7, which is listed in Appendix B of this report, ABB-CE performed a parametric evaluation and found that selecting either the upper curve or the lower curve for shear modulus variation was not necessarily conservative. However, in the letter dated April 15, 1992, ABB-CE asserted that the wide range of soil depths and shear wave velocities used in the System 80+ design covered the variations in response indicated by its parametric study.

The COL applicant must demonstrate that the assumptions made in the standard design regarding the variation of shear wave velocity and material damping are applicable to the site-specific design. ABB-CE included this in Section 2.5.4.7 of the final CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.4.8 Liquefaction Potential

ABB-CE states in CESSAR-DC Table 2.0-1 that sites with liquefaction potential at the site-specific SSE level will be excluded from consideration. The COL applicant must demonstrate that no liquefaction potential exists at the SSE level for the site for soils under and around all seismic Category I structures including Category I buried pipelines, tunnels, and electrical ducts.

The COL applicant must justify the selection of and the earthquake magnitude, duration, and the corresponding number of excitation cycles the soil properties used in the liquefaction potential evaluation (e.g., by submitting laboratory tests, field tests, and published data). In addition, the testing methods should be documented and subject to review and approval by the staff. ABB-CE included this in Section 2.5.4.8 of the final CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.4.9 Earthquake Design Basis

The earthquake design basis for the System 80+ design is discussed in CESSAR-DC Sections 2.5.2.6 and 2.5.2.7, and Appendix 2B. The staff's evaluations of these sections are provided in Sections 2.5.2 and 3.7.1 of this report.

2.5.4.10 Bearing Capacity

ABB-CE states in CESSAR-DC Table 2.0-1 that the minimum static bearing capacity of foundation material to support the superimposed loads from the structures is 59 ton/m² (12 ksf). The COL applicant must demonstrate that the site-specific soil-bearing capacity is equal to or exceeds the value in CESSAR-DC Table 2.0-1. The COL applicant should document the method used to establish the site-specific soil-bearing capacity and submit it for review and approval by the staff. ABB-CE included this in Section 2.5.4.10 of the final CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.5 Stability of Slopes

The COL applicant will be required to submit site-specific information on the stability of slopes. ABB-CE included this in Sections 2.5.5, 2.5.5.1, 2.5.5.2, and 2.5.5.3 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

2.5.6 Embankments and Dams

The COL applicant will be required to provide site-specific information on embankments and dams. ABB-CE included this in Sections 2.5.5, 2.5.5.1, 2.5.5.2, and

2.5.5.3 of CESSAR-DC. This is acceptable. This is included in COL Action Item 2.5-1.

Appendix 2A: Characteristics of Generic Soil Sites

The information included in this CESSAR-DC appendix is reviewed and evaluated in Section 2.5.2.5.2 of this report.

Appendix 2B: Characteristics of Selected Control Motions

The information included in this CESSAR-DC appendix is reviewed and evaluated in Section 2.5.2.5.1 of this report.

Appendix 2C: Strain-Compatible Modulus and Damping Values

The information included in this CESSAR-DC appendix is reviewed and evaluated in Section 2.5.2.5.2 of this report.

2.6 Site Parameter Envelope

Pursuant to 10 CFR 52.47(a)(1)(iii) an applicant seeking design certification is required to (1) submit site parameters postulated for the design and (2) analyze and evaluate the design in terms of these parameters. A list of site parameters for the System 80+ design is in CESSAR-DC Table 2.0-1. The staff's evaluation of the parameters in Table 2.0-1 is the following:

Groundwater - The site-specific maximum groundwater level is 0.6 m (2 ft) below grade; this is acceptable because this is a conservatively chosen ground water elevation and should cover the conditions prevalent at a majority of potential sites in the continental United States.

Maximum Flood (or Tsunami) Level - The site-specific maximum flood level is 0.3 m (1 ft) below grade which is consistent with the requirement of the EPRI Advanced Light Water Reactor Utility Requirements Document (URD), Volume II, "ALWR Evolutionary Plant," dated August 31, 1990. This is acceptable because the minimum design-basis flood level for preventing damage to seismic Category I SSCs is also at 0.3 m (1 ft) below plant grade. The maximum flood level is discussed in Section 3.4 of this report.

Precipitation (Maximum Rainfall Rate) - In its question (Q311.1 of Appendix B of this report), the staff questioned the appropriateness of using a maximum rainfall rate of 0.25 m/hr (10 in/hr). ABB-CE responded in its letter dated February 12, 1992, that the System 80+ design will change the maximum average rainfall to 0.49 m/km²/hr (19.4 in/mi²/hr) and 0.16 m/km²/5 min (6.2 in/mi²/5 min), to be consistent with Revision 1 of the EPRI URD. CESSAR-DC Table 2.0-1 states the latter values.

Site Characteristics

In NUREG-1242, "NRC Review of Electric Power Research Institutes Advanced Light Water Reactor Utility Requirements Document," dated August 1992, the staff's evaluation of the EPRI URD, the staff accepted the above rainfall rates but noted that a number of great lake area sites may not satisfy these parameters. Therefore, these rates are acceptable to the extent that the standard plant design will only be certified to the maximum rainfall rate described above.

Precipitation (Snow Load) - In its question (Q311.1 in Appendix B of this report), the staff questioned the appropriateness of using 244 kg/m² (50 psf) in CESSAR-DC Table 2.0-1 for the maximum snow load because this requirement may exclude some regions in the Great Lakes area and northeastern U.S., according to the ground snow load map formerly (Fig. 7) of American Society of Civil Engineers (ASCE) 7-88 (formerly ANSI A58.1-1982). ABB-CE responded that this 244 kg/m² (50 psf) value will be used in the design of System 80+ and a site-specific evaluation will be performed for plant sites with possible snow load in excess of 244 kg/m² (50 psf). The standard plant design will only be certified to a maximum snow load of 244 kg/m² (50 psf).

Design Temperatures - CESSAR-DC Table 2.0-1 specifies three categories of design temperatures: ambient, station service or emergency cooling water inlet, and condenser circulating water inlet. The water temperatures in the table are expressed in terms of the maximum and minimum temperatures for both the 1-percent exceedance probability and zero exceedance probability. In the DSER, the staff concluded that ABB-CE should clarify how the design temperatures listed in Table 2.0-1 will be used in the design of plant systems. This was DSER Open Item 2.6-1.

ABB-CE has specified the maximum and minimum design water temperatures for the plant systems used for cooling which are evaluated in Section 9.2 of this report. These maximum and minimum design water temperatures are consistent with the water temperatures in Table 2.0-1 of CESSAR-DC. This is acceptable. On this basis, DSER Open Item 2.6-1 is resolved.

Extreme Wind - The design-basis velocity for extreme winds differs from that for tornados because of the differences in recurrence internals and associated acceptance criteria such as design limits and loading combinations. CESSAR-DC Table 2.0-1 specifies basic windspeeds of 177 km/hr (110 mph) with importance factors of 1.0 and 1.11 for non-safety-related and safety-related structures, respectively. The design wind speed for safety-related structures is acceptable as discussed in Section 3.3.1 of this report.

Tornado - The System 80+ design uses missile Spectrum II (alternate spectrum) of SRP Section 3.5.1.4, "Missiles Generated by Natural Phenomena," which deviates from Table 1.2-6 of the URD, with missile Spectrum I. The approach is acceptable since both Spectra I and II meet the missile spectrum requirement of GDC 4 as given in SRP Section 3.5.1.4.

Soil Properties - The minimum soil-bearing capacity is 59 ton/m² (12 ksf) and sites with liquefaction potential at the site-specific SSE level are excluded. This information is reviewed and evaluated in Sections 2.5.4.8 and 2.5.4.10 of this report.

Seismology - The SSE and design ground response spectra are evaluated in Section 2.5 of this report. The response time history is evaluated in Section 3.7.1 of this report.

In the DSER, the staff concluded that the following site parameters should be in CESSAR-DC Table 2.0-1. This was DSER Open Item 2.6-2. This data has been added to the table with the EAB and LPZ distances increased; therefore, DSER Open Item 2.6-2 is resolved.

Aircraft Hazards

Plant to airport distance: 8 km (5 miles) < D < 16 km (10 miles) with annual operation less than 195 D² or

D > 16 km (10 miles) with an annual operation less than 390 D²

Plant to edge of military: D ≥ 8 km (5 miles) with an training routes with an annual operation less than or equal to 1000 flights

Plant to edge of Federal: D ≥ 3.2 km (2 miles) airway, holding pattern, or airport

Meteorology

Short-term dilution factor χ / Q 1.0×10^{-3} ;
EAB=800 meters (0.5 miles)

Long-term dilution factor χ / Q 2.2×10^{-5} ;
LPZ=3200 meters (2.0 miles)

The COL applicant should verify site-specific data to ensure that the data are bounded by those site envelope characteristics included in CESSAR-DC Table 2.0-1 and discussed above. The COL applicant must justify any sites

outside the design envelopes. ABB-CE included this in Section 2.0 of CESSAR-DC. This is acceptable. This is COL Action Item 2.0-1.

The meteorologic values in CESSAR-DC Table 2.3-1 is consistent with the CDM and CESSAR-DC Table 2.0-1. This resolves DSER Confirmatory Item 1.1-1.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS: Application of National Codes and Standards in Design Certification

Introduction

The staff's acceptance of national codes and standards for use in the design certification application is based on the latest edition of a given code or standard that has been reviewed and endorsed by the staff or endorsed with exceptions. The specific editions of the national codes and standards that are acceptable to the staff are part of the bases of the staff's finding in this safety evaluation report. The staff has reviewed and found acceptable those parts of codes and standards cited in this report, unless otherwise noted.

For design certification, ABB-CE should identify baseline-specific editions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereinafter referred to as "ASME Code") relating to in-service inspection (ISI) and in-service testing (IST). Each plant should meet all baseline ISI and IST requirements as specified in Sections 3.9.6 and 5.2.4 of this report. However, the requirements for the initial 10-year ISI and IST program must be determined by the edition of ASME Section XI in effect one year before the issuance of an operating license. Subsequent ISI and IST programs must be updated to the edition of Section XI in effect one year before the start of corresponding 10-year ISI and IST intervals. Because Section XI requirements evolve with the ASME Code, a licensee is required by 10 CFR 50.55a(f) and 50.55a(g) to meet the subsequent or new ISI and IST requirements to the extent practical, within the limitations of design, geometry, and materials of construction of the components. The regulations provide for the staff evaluation, upon written request, of new Section XI ISI and IST requirements determined by a licensee to be impractical for its facility. The staff has regulatory authority (10 CFR 50.55a(f) and (g)) to grant relief from specific impractical ISI or IST requirements and to impose alternative requirements, if appropriate.

The ABB-CE design is based on performing ISI and IST in accordance with 10 CFR 50.55a(f) and 50.55a(g). The combined license (COL) applicant should note the applicable ISI and IST code editions and submit the corresponding ISI and IST programs for staff review and approval in accordance with 10 CFR 50.55a(f) and 50.55a(g). This is included in COL Action Items 1-2, 3.9-1, and 5.5.

However, pre-service inspection (PSI) requirements are related to the design, which is based on specific editions of the ASME Code identified in the design certification. Details related to PSI are discussed in Sections 5.2.4 and 6.6 of this report. Because the PSI requirements can be incorporated into the component design, 10 CFR 50.55a(g)

does not have provisions for relief from PSI requirements, and ABB-CE has not noted any deviations from the 1989 Edition of the Code.

Detailed discussions on ISI and IST are in Sections 3.9.6, 5.2.4, and 6.6 of this report.

In the draft safety evaluation report (DSER), the staff noted that ABB-CE should verify that the CESSAR-DC identifies specific editions of all national codes and standards. This was DSER Open Item 3-1. CESSAR-DC Table 1.8-6 identifies the specific editions of the applicable national codes and standards, including the 1989 Edition of the ASME Code and addenda, for the System 80+ design. Relevant parts of this specific edition of the ASME Code have been endorsed by 10 CFR 50.55a, and are, therefore, acceptable. On this basis, DSER Open Item 3-1 resolved.

By DSER COL Action Item 3.9.3.1-1, the staff requested that the COL applicant verify the edition of the ASME Code to be used in its design in accordance with Chapter 3 of this report. The staff's position is that the COL applicant may propose editions of national codes and standards that are different from those specified identified in CESSAR-DC Table 1.8-6, but the COL applicant must submit information containing a summary of its evaluation for each change and provide the bases in a "50.59-like" determination. As set forth below, CESSAR-DC Sections 3.9.3 and 5.2.1.1 state that the site-specific safety analysis report (SAR) will provide information on the specific editions of the ASME Code in the site-specific design and that ASME Code editions and addenda other than those specified in CESSAR-DC Table 1.8-6 shall have been endorsed by 10 CFR 50.55a. This is acceptable. This is included in COL Action Item 1-2.

3.1 General

The staff reviewed the information in CESSAR-DC Section 3.1 to verify that the System 80+ standard design will meet the general design criteria (GDC) of Appendix A to 10 CFR Part 50.

3.1.1 Elimination of Operating Basis Earthquake (OBE) from Design Consideration

Appendix A to 10 CFR Part 100 requires, in part, that all structures, systems, and components (SSCs) of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress

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and deformation limits when subject to an OBE. Changes to Appendix A to 10 CFR Part 100 are being proposed to redefine the OBE such that the OBE criterion can be satisfied without the need to perform explicit design and response analyses, if the OBE is equal to or less than one-third of the safe-shutdown earthquake (SSE). The proposed amendment to 10 CFR Part 100, Appendix A, would also allow, as an option, that the OBE be eliminated from design certification when the OBE is established at less than or equal to one-third of the SSE. In this manner, the OBE serves the function as an inspection level earthquake below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damages. In addition, the staff notes that Appendix A to 10 CFR Part 100 requires that the maximum vibratory ground acceleration of the OBE be at least one-half the maximum vibratory ground acceleration of the SSE. Therefore, deviations from this regulation must be evaluated to ensure that 10 CFR 50.12 exemption criteria are satisfied when the OBE is eliminated from design analysis and is redefined only as an inspection level earthquake with the maximum vibratory ground acceleration at one-third of that of the SSE.

Special circumstances exist for granting these exemptions from the requirements of Appendix A to Part 100 pursuant to 10 CFR 50.12. The purpose of designing SSCs necessary for continued operation without undue risk to the health and safety of the public to withstand an OBE is to ensure that these SSCs remain functional and within applicable stress and deformation limits when subjected to the effects of the OBE vibratory ground motion. However, Appendix A to Part 100 also requires that these SSCs be designed to withstand the SSE and remain functional. Thus, when these SSCs are designed to remain functional for the SSE, they will also remain functional at a lesser earthquake level (one-third the SSE) provided that all design functions of the OBE are accounted for. The basis for selecting one-third the SSE as the earthquake level at which the plant will be required to shut down and be inspected for damage was that at this level the likelihood of damage and the frequency of earthquakes occurring was judged to be low based on actual earthquake experience. It should be noted that certain design functions had been verified only for the OBE loads in the past. These design functions were the evaluations of (1) fatigue damage caused by earthquake cycles and (2) relative seismic anchor motions (SAMs) in piping systems. With the elimination of the OBE from design, these design functions would not explicitly have been verified. Consequently, for the System 80+ these design functions will be verified in conjunction with the SSE using applicable stress and deformation limits as described below.

As described more fully below, the special circumstances described by 10 CFR 50.12(a)(2)(ii) exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose of the rule because ABB-CE has proposed acceptable alternative analysis methods that accomplish the intent of the regulation. On this basis, the staff concludes that the exemption is justified because the alternative analyses performed for the SSE and the need to inspect the plant following an earthquake at or above one-third the SSE accomplish the design objectives of the OBE design analyses.

Background

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff requested the Commission's approval to decouple the level of the OBE ground motion from that of the SSE. The Commission approved the staff's position in its staff requirements memorandum (SRM) of June 26, 1990. In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the staff further requested that the Commission approve eliminating the OBE from the design of SSCs in both evolutionary and passive advanced reactors designs. The Commission approved this staff position in its SRM dated July 21, 1993. The elimination of the OBE from design was requested by the Electric Power Research Institute (EPRI) and also recommended by the Advisory Committee on Reactor Safeguards (ACRS) in its letter of April 26, 1990.

In SECY-93-087, the staff examined the safety impact of eliminating the OBE as it pertains to civil structures, piping systems, and equipment seismic qualification. The staff also made several recommendations to ensure that eliminating the OBE would not result in a significant decrease in the overall plant safety margin. In its SRM of July 21, 1993, the Commission approved and agreed with the following staff positions:

- Use two SSE events with 10 maximum stress cycles per event to account for earthquake cycles in the fatigue analyses until the staff issues new guidance; alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of Institute of Electrical and Electronics Engineers (IEEE) 344-1987.
- Consider the effects of anchor displacement in the piping caused by an SSE with the Service Level D limit.

- Eliminate the OBE from the design of SSCs. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations.
- The mechanistic pipe break and high-energy leakage crack locations determined by the piping high stress (without earthquake loads) and fatigue locations (with earthquake loads) may be used for equipment environmental qualification and compartment pressurization purposes.
- With the elimination of the OBE, two alternatives exist that will essentially maintain the requirements provided in IEEE 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE event. Of these alternatives, the equipment should be qualified with five one-half SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance with Appendix D of IEEE 344-1987 when followed by one full SSE.
- Continue to use the OBE as a threshold criterion for conducting inspection following an earthquake event.

The following sections present the staff's evaluation of the commitments specified in the CESSAR-DC to ensure that appropriate measures and adequate safety margins are maintained when the OBE is eliminated from the design. The sections evaluate (1) ASME Code Class 1, 2, and 3 components and core support structures, (2) concrete and steel structures, (3) equipment seismic qualification, and (4) pre-earthquake planning and post-earthquake operator actions.

ASME Code Class 1, 2, and 3 Components and Core Support Structures

The dynamic analysis methods for seismic analyses of ASME Code Class 1, 2, and 3 components and core support structures in the System 80+ are described in the CESSAR-DC as approved by the NRC staff in Sections 3.9.2 and 3.12.3 of this report. The loads and load combinations used for evaluating ASME Code Class 1, 2, and 3 components and core support structures are provided in the CESSAR-DC and are also discussed in Sections 3.9.3.1, 3.12.5.3, and 3.12.5.15 of this report. Conformance to existing staff guidelines that ensure the operability of safety-related pumps and valves under SSE loading conditions are discussed in Section 3.9.3.1 of this report. The functionality of the supported system has also been taken into account. As specified in Regulatory Guide (RG) 1.124, "Service Limits and Loading Combinations

for Class 1 Linear-Type Components Supports," Revision 1, to ensure that systems — whose normal function is to prevent or mitigate consequences of events associated with the SSE — will operate adequately regardless of plant condition, the Code Level B service limits of Subsection NF or other justifiable limits approved by the staff have been used.

The elimination of the OBE from the analysis and design of ASME Code Class 1, 2, and 3 components and core support structures requires all current OBE design-related checks to be performed for the SSE. With regard to primary stress effects (seismic inertial stresses), the elimination of the OBE from Service Level B analysis could have a potential impact on design in those cases where the load combination includes other dynamic loadings (i.e., operational transients) in Service Level B but not in Service Level D. The staff explored this possibility specifically for piping systems and found that when the OBE is established at one-third of the SSE, the load combinations. Although cases have been found where the OBE controlled the design, they were typically found when other normal operating loads were large and did not result in a different hardware configuration. Therefore, for primary stresses in piping systems, the staff finds that eliminating the OBE from piping stress load combinations will not reduce existing safety margins because the load combination with the SSE loading is generally controlling.

For cyclic and secondary stress effects (e.g., fatigue and SAM), the elimination of the OBE would have a direct impact on the current methods used to evaluate the adequacy of piping design. Because the cyclic (fatigue) effects of earthquake-induced motions in piping systems and the relative motion effects of piping anchored to equipment and structures at various elevations are currently evaluated only for OBE loadings, the elimination of the OBE from the load combination could lead to uncertainty concerning how these effects should be evaluated. The following paragraphs discuss the staff's evaluation of the System 80+ guidelines, presented in the CESSAR-DC, for treating these effects.

- (1) Fatigue — In order to ensure adequate design considerations for the fatigue effects of earthquake cycles, it is necessary to establish a bounding load definition and number of earthquake cycles to account for the more frequent occurrences of lesser earthquakes and their aftershocks. In CESSAR-DC Section 3.7.3.2, ABB-CE uses a cyclic load basis for fatigue analysis of earthquake loading for ASME Code Class 1, 2, and 3 components and core support structures equal to two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). This is

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equivalent to the cyclic load basis of one SSE and five OBE events as currently recommended in the Standard Review Plan (SRP) (NUREG-0800) Section 3.9.2. Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE 344-1987.

- (2) Seismic Anchor Motion (SAM) — For the System 80+, the effects of displacement-limited SAM due to an SSE are evaluated for safety-related ASME Code Class 1, 2, and 3 components and component supports to ensure their functionality during and following an SSE. The SAM effects include (but are not limited to) relative displacements of piping between building floors and slabs, at equipment nozzles, at piping penetrations, and at connections of small-diameter piping to large-diameter piping.

For piping systems, the effects of SAMs due to an SSE are combined with the effects of other normal operational loadings that might occur concurrently as specified in Table 3.9.2 of the CESSAR-DC.

- (3) Piping Stress Limits — For ASME Code Class 1, 2, and 3 piping, the design requirements in the 1989 Edition of the ASME Code, Section III, Subsections NB, NC, and ND shall be met. In addition, the following changes and additions to paragraphs NB-3650, NC-3650, and ND-3650 will be used for piping systems when the OBE is eliminated from the design.

All equation numbers and terms used below are identified and defined in ASME Code, Section III, unless otherwise noted.

ASME Code Class 1 Piping Stress Limits

- (a) For primary stress evaluation (NB-3654.2), earthquake loads are not required to be evaluated for consideration of Level B Service Limits for Equation (Eq.) (9).
- (b) For satisfaction of primary plus secondary stress intensity range (NB-3653.1), in Eq. (10), M_i shall be either (1) the resultant range of all loads considering one-half the range of the SSE or (2) the resultant range of moment due to the full range of the SSE alone, whichever is greater. The use of the SSE is intended to provide a

bounding design for the cumulative effects of earthquakes of a lesser magnitude and is therefore to be included in consideration of Level B Service Limits for Eq. (10). A reduced range (with an equivalent number of fractional vibratory peak cycles) of the SSE moment may be used for consideration of Level B Service Limits (but with a range not less than one-third of the maximum SSE moment range).

- (c) For satisfaction of peak stress intensity (NB-3653.2), the load sets developed in NB-3653.1 based on the above Position 3(b) will be used in calculating the peak stress intensity, S_p , and the alternating stress intensity, S_{alt} , for evaluating the fatigue effects and cumulative damage.
- (d) For simplified elastic-plastic discontinuity analysis (NB-3653.6), if Eq. (10) cannot be satisfied for all pairs of load sets, then the alternative analysis as described in NB-3653.6 will be followed. For treatment of SAM moments, the following condition shall be satisfied in consideration of Level D Service Limits:

$$S_{SAM} = C_2 \frac{D_o}{2I} (M_i^* + M_i^{**}) \leq 6.0 S_m$$

where: S_{SAM} is the nominal value of SAM stress

M_i^* is the same as M_i^* in Eq. (12)

M_i^{**} is the same as M_i in Eq. (10), except that it includes only moments due to SAM displacements caused by an SSE

The combined moment range ($M_i^* + M_i^{**}$) shall be either (1) the resultant range of thermal expansion and thermal anchor movements plus one-half the range of the SSE anchor motion or (2) the resultant range of moment due to the full range of the SSE anchor motion alone, whichever is greater.

ASME Code Class 2 and 3 Piping Stress Limits

- (a) For consideration of occasional loads (NC/ND-3653.1), earthquake loads (i.e., inertia and SAM) are not required for satisfying Level B Service Limits for Eq. (9).

- (b) For consideration of thermal expansion or secondary stresses (NC/ND-3653.2), M_c in Eq. (10) is not required to include the moment effects of SAMs due to an earthquake.
- (c) For consideration of secondary stresses in Level D Service Limit (NC/ND-3655), the following condition will be satisfied:

$$S_i = i \frac{M_c^* + M_c}{z} \leq 3.0 S_h \quad \text{Eq. (10b)}$$

where: M_c^* is the range of moments due to SAMs due to an SSE M_c is the range of moments due to thermal expansion

The combined moment range ($M_c^* + M_c$) shall be either (1) the resultant range of moments due to thermal expansion plus one-half of the range of moments due to the SSE anchor motions, or (2) the resultant range of moment due to the full range of the SSE anchor motion alone, whichever is greater.

Upon reviewing the supplemental criteria above to be used when the OBE is eliminated from the design of piping systems, the staff finds that the criteria strive to maintain the existing design margins of the ASME Code, Section III although some criteria appear to be more stringent and others more relaxed. The net effect results in safety margins equivalent to that of the 1989 Edition of the ASME Code, Section III, rules. This gives a more controlled check of piping system stresses in those areas where actual failures of piping systems due to seismic loadings have occurred. The staff concludes that the piping criteria for the System 80+ meet the staff recommendations in SECY-93-087 for considering earthquake cycles in fatigue analyses and for evaluating the effects of anchor displacements in the piping caused by an SSE and are, thus, acceptable.

- (4) **Pipe Break Postulation Without OBE** — It is recognized that pipe rupture is a rare event which might occur only under unanticipated conditions of design, construction, or operational errors or unanticipated loads or unanticipated corrosive environments. The staff's observation of actual piping failures have found that they generally occur at high stress and fatigue locations, such as at the terminal ends of a piping system at its connection to component nozzles. Currently, in accordance with SRP

(NUREG-0800) Section 3.6.2, Revision 2 dated June 1987, pipe breaks are postulated in high energy piping at locations of high stress and high fatigue usage factor. The load combination used in calculating the high stress and usage factor includes normal and upset load conditions (i.e., pressure, weight, thermal, OBE, and other operational transient loadings).

From a historical viewpoint, the criteria for postulating high-energy breaks at specified locations were first introduced in the early 1970s. The basis for the mechanistic approach for selecting pipe break locations was derived from the premise that although pipe breaks could result from random events induced by unanticipated conditions, the failure mechanism and the expected location of failure would likely be caused by local conditions of high stress or high fatigue in the piping. In order to ensure that a sufficient number of pipe breaks would be postulated, breaks were recommended to be postulated for a wide spectrum of events to envelop the uncertainties of unanticipated failure mechanisms. Breaks were postulated at terminal ends of the piping, at high-stress and high-fatigue locations, and as a minimum at two additional intermediate locations when the stresses were below the high stress threshold limit. The resulting criteria which were incorporated in SRP Section 3.6.2 resulted in many postulated pipe break locations and caused the installation of numerous pipe rupture mitigation devices in nuclear plants.

In the mid-1980s, the Nuclear Regulatory Commission's (NRC's) Executive Director for Operations (EDO) initiated a comprehensive review of nuclear power plant piping to identify areas where changes to the piping requirements could improve the licensing process as well as the safety and reliability of nuclear power plants. The NRC's Piping Review Committee (PRC) in an integrated effort with the nuclear industry under the Pressure Vessel Research Committee (PVRC) conducted a comprehensive study of piping criteria including the mechanistic pipe break postulation guidelines. The PRC found that when an excessive number of pipe rupture mitigation devices (i.e., pipe whip restraints and jet impingement shields) are installed on high energy piping systems, the potential exists for piping systems to be overly constrained. This condition was found in several nuclear plants in which massive pipe restraints adversely affected the ability of the high-temperature piping to freely expand during normal plant operation. The PRC also found through numerous dynamic tests and field observations of non-seismically designed piping systems that had undergone high-seismic loadings that butt-welded piping possesses an inherent ability to withstand large seismic inertial loadings without failure.

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As a result of the PRC's effort, the NRC staff recognized that the mechanistic pipe rupture criteria for selecting locations of pipe breaks resulted in an excessive number of pipe rupture mitigation devices that could hinder the normal operation of the plant and might not contribute significantly to the overall safety of the plant. Accordingly, the SRP was revised to reduce the number of postulated pipe breaks by (1) eliminating the need to postulate pipe breaks at the two arbitrary intermediate locations and (2) providing a leak-before-break (LBB) approach in lieu of postulating pipe breaks when the system and material specific information is adequate to justify its application.

On the basis of recent dynamic pipe tests conducted by EPRI and NRC in the Piping and Fitting Dynamic Reliability Program, it has been demonstrated that the piping can withstand seismic inertial loadings higher than an SSE without rupturing. Thus, the likelihood of a pipe break in a seismically designed piping system due to an earthquake magnitude of one-third SSE is remote. Operating experience has shown that pipe breaks are more likely to occur under conditions caused by normal operation (e.g., erosion-corrosion, thermal constraint, fatigue, and operational transients).

On the basis of the above discussion, the staff concludes that no replacement earthquake loading should be used to establish postulated pipe break and leakage crack locations due to high stresses in the pipe. Instead, the criteria for postulating pipe breaks and leakage cracks in seismically designed, high- and moderate-energy piping systems should be based on factors attributed to normal and operational transients only. CESSAR-DC Sections 3.6.1 and 3.6.2 conform to the above staff's position of pipe break postulation. However, for establishing pipe breaks and leakage cracks due to fatigue effects, the calculation of the cumulative usage factor will continue to include seismic cyclic effects. The staff's revised criteria for pipe break postulation are given below. The revised criteria are intended to ensure that breaks and leakage cracks are postulated to occur at the most likely locations and to reduce the number of pipe rupture mitigation devices (e.g., pipe whip restraints and jet impingement shields) that might hinder plant operation without giving a compensatory level of safety.

Consistent with the above staff finding, the guidelines in SRP Section 3.6.2, Branch Technical Position (BTP) Mechanical Engineering Branch (MEB) 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," have been revised for the System 80+ as follows:

B.1.b.(1).(a): Footnote 2 should read, "For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification (excluding earthquake loads)."

B.1.b.(1).(d): "The maximum stress as calculated by the sum of Eqs. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits have been specified in the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) excluding earthquake loads should not exceed $0.8(1.8 S_h + S_A)$."

The System 80+ criteria are consistent with the above staff position for postulating pipe breaks and cracks. The staff concludes that the criteria meet the staff recommendations in SECY-93-087 and thus, are acceptable.

Concrete and Steel Structures

The current design practice for considering OBE and SSE ground motion effects in the seismic design of nuclear plant structures was established in the 1960s with conceptual goals of (1) maintaining continued plant operation without damage to the structures for OBE level earthquakes and (2) ensuring safe shutdown of plant and maintaining the plant in a safe-shutdown condition during and after the occurrence of an SSE. To achieve these goals, the structural responses are kept at or below the material yield stresses to preclude the onset of plastic deformation for load combinations due to accident conditions plus the SSE. For load combinations due to operating conditions plus the OBE, stresses are limited at one-half to five-eighths yield stress. The current load combinations in SRP Section 3.8 were developed from the above design philosophy.

In the design of the containment, the staff reviewed the extent to which the elimination of the OBE from the load combinations would lead to a reduction of the safety margin. An examination of the nuclear structural design practice and the SRP Sections 3.8.1, 3.8.2, and 3.8.3 load combination equations, however, shows that the major dynamic load for the overall design of structures is either the OBE or the SSE. For Category I steel and concrete structures, the staff's guidance on load combinations is in SRP Section 3.8.4. The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures

although there may be specific cases where the load combinations with the OBE control. Similar to the piping design, an examination of the pertinent load combinations for concrete structures should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads, except those that produce local effects that are handled by local reinforcement details. Therefore, it can be concluded that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

For the System 80+, the following criteria for structures are used to ensure that when the OBE is eliminated from design, the structures will continue to be designed appropriately for earthquake effects.

- (1) SSE Relative Displacements Between Structures — In CESSAR-DC Section 3.7.2, the seismic response (building displacements, structural member forces, floor response spectra (FRS), etc.) of the nuclear island (NI) structures is discussed. ABB-CE has considered the through-soil, structure-to-structure interaction effect under SSE loading in the analyses of System 80+ structures including the NI structures, component cooling water (CCW) heat exchanger (Hx) structures, diesel fuel storage structures (DFSSs), radwaste building, and turbine building. Therefore, the staff concludes that the effects of through-soil, structure-to-structure interaction under SSE loadings for all structures housing seismically-designed piping have been adequately considered under SSE loadings to establish the relative displacements between buildings (seismic anchor movement for piping systems).
- (2) Seismic Instrumentation — CESSAR-DC Sections 3.7.4 states that seismic instrumentation will be installed so that prompt action can be taken when the response spectra level and the cumulative absolute velocity (CAV) experienced at the site exceed the shutdown level. It also gives information on pre-earthquake planning and post-earthquake operator action. The acceptability review of seismic instrumentation is discussed in Section 3.7.4 of this report.
- (3) Use of RGs 1.143 and 1.27 — The staff guidelines in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled

Nuclear Power Plants," Revision 1, and in RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 2, provide for a seismic design of radwaste buildings and ultimate heat sink features based on the OBE. With the elimination of the OBE, ABB-CE proposed that these structures and features be designed to withstand the SSE. The structural design criteria using the SSE loading use the corresponding loads and load combinations provided in SRP Section 3.8.4. The staff finds that designing these structures and features to the SSE will give a conservative design comparable to that given in the regulatory guides and is thus acceptable. Alternate methods to ensure the seismic adequacy of these structures and features will be reviewed by the staff on a case-by-case basis.

Equipment Seismic Qualification

The proposed elimination of OBE from explicit design consideration affects different aspects of equipment qualification in different manners. In the area of equipment qualification, the requirements in the regulations (10 CFR Parts 50 and 100) are interpreted by the staff through SRP Section 3.10, which deals with seismic and dynamic qualification of mechanical and electrical equipment.

When the equipment is qualified by analysis, the acceptance criteria are derived from the ASME Code. The effect of eliminating the OBE on equipment qualification by analysis should be negligible. It is well known that mechanical equipment such as pumps and valves are, in general, seismically rugged when adequately anchored, and that their operability limits are generally established through maximum permissible moments and forces or tolerance limits based on available clearances that are controlled by the SSE rather than the OBE. Therefore, for mechanical equipment, elimination of OBE from qualification analysis should not reduce any margin. Also, some electrical equipment is allowed to be qualified by analysis which requires demonstration that five OBE events followed by one SSE event do not cause its failure to perform its safety function. With the elimination of OBE, analysis checks for fatigue effects may be done at a fraction of the SSE (e.g., 50 cycles at one-half of the SSE peak amplitude or 150 cycles at one-third of the SSE peak amplitude).

When equipment for seismic loadings is qualified by analysis, testing, or a combination of both, the staff recommends the use of IEEE 344-1987 as endorsed in RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Revision 2. For analysis, the selection of the level of service limit for

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different loading combinations shall ensure the functionality of the equipment during and following a SSE. For testing, IEEE 344-1987 has detailed requirements for performing seismic qualification using five OBE events followed by an SSE event. Where complex mathematical models are based solely on calculated structural parameters, verification testing should be done.

With the elimination of the OBE, and in order to maintain the equivalent qualification requirements in IEEE 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE event, the staff recommended, in SECY-93-087, that equipment be qualified with five half-SSE events followed by one full SSE event. Alternatively, the staff recommended a number of fractional peak cycles equivalent to the maximum peak cycles for five half-SSE events may be used in accordance with Appendix D of IEEE 344-1987 when followed by one full SSE. In CESSAR-DC Section 3.7.3.2 and Section 3.10, ABB-CE commits to the above staff recommendations as stated in SECY-93-087. Thus, the CESSAR-DC criteria for equipment seismic qualification, when the OBE is eliminated from design, are acceptable.

Pre-Earthquake Planning and Post-Earthquake Operator Actions

The design certification of the System 80+ using a single-earthquake (SSE) design is predicated on the adequacy of pre-earthquake planning and post-earthquake inspections for damage that are to be implemented by the COL applicant. The COL applicant will be required to submit to the NRC as a part of its application the procedures it plans to use for pre-earthquake planning and post-earthquake actions. For design certification, the NRC staff reviewed the criteria developed in EPRI Reports NP-5930, "A Criterion for Determining Exceedance of the Operating Basis Earthquake," July 1988, NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," December 1989, and TR-100082, "Standardization of the Cumulative Absolute Velocity," December 1991, for evaluating the need to shut down the plant following an earthquake and the commitments for the System 80+ for ensuring that these actions can be taken as provided for in CESSAR-DC Section 3.7.4.

EPRI NP-5930

The staff's review of EPRI Report NP-5930 finds that it is adequate for, and may be used by, a COL applicant with the following exceptions:

- (1) A free-field instrument must be used for determining the CAV and the spectral acceleration level.

- (2) The response spectrum check is as follows:

The 5-percent damped ground response spectrum for the earthquake motion at the site exceeds (1) one-third the corresponding SSE response spectral acceleration between 2 and 10 hertz (Hz), or it exceeds an acceleration of 0.20 g between 2 and 10 Hz whichever is greater or (2) one-third the corresponding SSE response spectral velocity between 1 and 2 Hz or a velocity of 15 cm/second (6 in/second) between 1 and 2 Hz, whichever is greater.

- (3) The licensee shall shut down the plant if an earthquake results in a simultaneous exceedance of the 5-percent damped ground response spectrum enumerated in item 2 and a CAV exceedance of 0.16 g-sec for any one frequency on any one component of the free field ground motion. The CAV shall be determined in accordance with EPRI Report TR-100082. Also, any evidence of significant damage observed during the plant walkdown in accordance with the EPRI Report NC-6695 recommendations shall be sufficient cause for plant shut-down.
- (4) The instrumentation installed at the nuclear power plant shall be capable of on-line digital recording of all three components of the ground motion and of converting the recorded (digital) signal into the standardized CAV and the 5-percent damped response spectrum. The digitizing rate of the time history of the ground motions shall be at least 200 samples per second and the band-width shall be at least from 0.20 Hz to 50 Hz. The pre-event memory of the instrument shall be sufficient to record the onset of the earthquake.
- (5) The system must be capable of routinely calibrating the response spectrum check of 0.20 g. Also, the CAV of 0.16 g-sec should be calibrated with a copy of the October 1987 earthquake in Whittier, California or an equivalent calibration record provided for this purpose by the manufacturer of the instrumentation. In the event that an actual earthquake has been recorded at the plant site, the above calibration shall be done to demonstrate that the system was functioning properly at the time of the earthquake.

EPRI NC-6695

The staff's review of the EPRI Report NC-6695 finds that it may be used by the COL applicant with the following exceptions:

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- Section 3.1, Short-Term Actions

- Item 3, "Evaluation of Ground Motion Records"

There is a time limitation of 4 hours within which the licensee shall determine if the shutdown criterion has been exceeded. After an earthquake has been recorded at the site, the licensee shall submit a response spectrum calibration record and CAV calibration record to demonstrate that the system was functioning properly.

- Item 4, "Decision on Shutdown"

Exceedance of the EPRI criterion as amended by the NRC or observed evidence of significant damage as defined by EPRI NC-6695 shall constitute a condition for mandatory shutdown, unless conditions prevent the licensee from accomplishing an orderly shutdown without jeopardizing the health and safety of the public.

- Add Item 7, "Documentation"

The licensee shall record the chronology of events and control room problems while the earthquake evaluation is in progress.

- Section 4.3, "Guidelines" (p. 4-3)

Because earthquake-induced vibration of the reactor vessel could lead to changes in NFs a prompt check of the NF monitoring instruments shall be made to indicate if the reactor is stable. Therefore, this check should be added to the checks listed in this section.

- Section 4.3.4, "Pre-Shutdown Inspection"

Exceeding the EPRI criterion or evidence of significant damage should constitute a condition for mandatory plant shutdown, as the staff stated in its recommendation for Section 3.1, item 4, "Decision on Shutdown."

- Section 4.3.4.1, "Safe-Shutdown Equipment" (p. 4-7)

In addition to the safe shutdown systems on this list, containment integrity must be maintained following an earthquake. Since the containment isolation valves (CIVs) may have malfunctioned during the earthquake, inspection of the containment isolation system is necessary to assure continued containment integrity.

Conclusions

On the basis of the changes to the existing seismic design criteria discussed above, the staff concludes that eliminating the OBE from the design of SSCs in the System 80+

standard plant will not reduce the level of safety given in current regulatory guidelines for seismic design. On the contrary, the staff finds that the changes provide an enhancement to safety by refocusing current design requirements to emphasize those areas where failure modes are more likely to occur and by precluding the need for seismic design requirements that do not significantly contribute to the overall safety of the plant.

On the basis of its review of relevant CESSAR-DC sections reflecting the above positions, the staff concludes that, as an exemption from the regulation in accordance with the criteria described by 10 CFR 50.12(a)(2)(ii), the elimination of the OBE from the design of SSCs in the System 80+ standard plant meets the Commission-approved staff recommendations in SECY-93-087 and is, therefore, acceptable.

3.2 Classification of SSCs

By DSER Open Item 3.2-1, the staff noted that ABB-CE should submit the final version of the distribution systems design guide (DSDG). As the result of the receipt of a final DSDG dated January 15, 1993, for staff review, DSER Open Item 3.2-1 is resolved. Information necessary for the staff's safety evaluation was extracted by ABB-CE from the DSDG, modified, or clarified as necessary and subsequently included in Appendix 3.9A to the CESSAR-DC by Amendment P. The staff's review of Appendix 3.9A is discussed in Section 3.12 of this report.

3.2.1 Seismic Classification

GDC 2, "Design Bases for Protection Against Natural Phenomena," in part, requires that nuclear power plant SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Certain of these features are necessary to ensure (1) the integrity of the reactor coolant pressure boundary (RCPB), (2) the capability to shut down the reactor and maintain it in a safe-shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures that are comparable to the guidelines in 10 CFR Part 100. The earthquake for which these safety-related plant features are designed is defined as the SSE in 10 CFR Part 100, Appendix A. The SSE is based on an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which SSCs are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated seismic Category I in RG 1.29, "Seismic Design Classification," Revision 3. The staff reviewed the CESSAR-DC in

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accordance with SRP Section 3.2.1, which references RG 1.29.

The SSCs of the System 80+ standard design, which includes the reactor building (RB) and the nuclear annex (NA), that are required to be designed to withstand the effects of an SSE and remain functional are listed in CESSAR-DC Table 3.2-1. This table, in part, lists major components in fluid systems, mechanical systems, and associated structures designated as seismic Category I.

In its request for additional information (RAI) question (Q)210.10, the staff noted that ABB-CE did not clearly classify the seismic category for piping and component supports in the CESSAR-DC. In response, ABB-CE modified CESSAR-DC Section 3.2.1, in Amendment K, to state that piping supports and component supports are of the same seismic category as the piping and components to which they apply. This is acceptable because piping and component supports shall be classified in accordance with RG 1.29 to the same seismic category of their associated piping and components. On this basis, DSER Confirmatory Item 3.2.1-1 is resolved.

CESSAR-DC Sections 3.2.1, 3.7.2.8, and 3.7.3.13 state that equipment, structures, and piping in the System 80+ design that are non-seismic Category I but that could degrade or cause the failure of seismic Category I items if their structural integrity is lost, are analyzed and designed to ensure that their integrity is maintained under seismic loading from the SSE. By DSER Open Item 3.2.1-1, the staff requested that ABB-CE clarify the use of the term "restrained as required." Amendment N to CESSAR-DC Sections 3.7.3.13(A) and 3.7.3.13(C) stated that the attached non-Category I piping, up to the first anchor beyond the interface, is designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the Category I piping. These commitments are in conformance with RG 1.29, and thus, acceptable. On this basis, DSER Open Item 3.2.1-1 is resolved.

On the basis of its review of CESSAR-DC Section 3.2.1, the staff concludes that the SSCs important to safety that must be designed to withstand the effects of an SSE and remain functional are properly classified as seismic Category I in accordance with RG 1.29 and the design constitutes an acceptable basis for satisfying the requirements of GDC 2 and is, therefore, acceptable.

3.2.2 System Quality Group Classification

GDC 1, "Quality Standards and Records," in 10 CFR Part 50, Appendix A, requires that nuclear powerplant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement is applicable to both pressure-retaining and non-pressure-retaining SSCs that are part of the RCPB and other systems important to safety, where reliance is placed on these systems to (1) prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB, (2) permit shutdown of the reactor and maintain it in a safe-shutdown condition, and (3) retain radioactive material.

The staff reviewed the CESSAR-DC for compliance with GDC 1 in accordance with SRP Section 3.2.2, "System Quality Group Classification." SRP Section 3.2.2 references RG 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 3, as the principal document used in the staff review for identifying, on a functional basis, the pressure-retaining components of those systems important to safety as NRC Quality Group A, B, C, or D. Those ASME Code, Section III, Class 1 components that are part of the RCPB are identified in 10 CFR 50.55a. Conformance of these RCPB components to 10 CFR 50.55a is discussed in Section 5.2.1.1 of this report. These RCPB components are designated in RG 1.26 as Quality Group A. Certain other RCPB components that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified as Quality Group B in accordance with RG 1.26.

ABB-CE used American Nuclear Society (ANS) Safety Class 1, 2, 3, and non-nuclear safety (NNS) as defined in American National Standards Institute (ANSI)/ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," for the classification of system components as an alternative method of meeting RG 1.26. CESSAR-DC Section 3.2.2 and Table 3.2-3 correlate (1) CESSAR-DC Safety Class 1, 2, 3, and NNS of ANSI/ANS 51.1, (2) Quality Group A, B, C, and D of RG 1.26, and (3) ASME Code Section III classes. The relationship between the three methods of classification in the CESSAR-DC is summarized in the following table. Note that this table is only applicable to pressure-retaining components.

NRC Quality Group	System 80+ Safety Class	ASME Code Section III Safety Class
A	1	1
B	2	2
C	3	3
D	NNS	-

All pressure-retaining components classified as Quality Group A, B, or C are constructed in accordance with ASME Code, Section III, Class 1, 2, or 3 rules, respectively. Construction, as defined in ASME Code, Section III, Article NCA-1110, and as used herein, is an all-inclusive term comprising materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components. Components classified as Quality Group D are designed to the applicable standards noted in CESSAR-DC Table 3.2-3.

This table (above) is acceptable for defining the relationship between the three methods of classification for pressure-retaining components. However, the staff has not completely endorsed ANSI/ANS 51.1-1983 and cannot rely on the safety classifications in this document in determining the acceptability of non-pressure-retaining SSCs. Therefore, for non-pressure-retaining components, the staff's review of CESSAR-DC Table 3.2-1 concentrates on an evaluation of quality assurance (QA) in accordance with 10 CFR Part 50 (Appendix B) and seismic classifications.

The staff also relies on its review of system piping and instrumentation diagrams (P&IDs) to determine the appropriate classification boundaries for pressure-retaining components. The P&IDs that were available in the CESSAR-DC when the staff reviewed it were generally acceptable and properly classified pressure-retaining components in accordance with RG 1.26. By letter dated April 15, 1992 (LD-92-049), ABB-CE transmitted revised P&IDs and flow diagrams for auxiliary systems including the chemical and volume and control system (CVCS). The staff reviewed the revised P&IDs for the CVCS, which were incorporated into Amendment K to CESSAR-DC Figure 9.3.4-1, and found them consistent with CESSAR-DC Tables 3.2-1 and 3.2-2. On this basis, DSER Open Item 3.2.2-1 is resolved.

By DSER Open Item 3.2.2-2, the staff noted that discrepancies were found in safety classifications for the CVCS,

and that ABB-CE's response related to generic safety issue (GSI)-23 (reactor coolant pump (RCP) seal integrity issue) was inadequate. In Amendment K to CESSAR-DC, ABB-CE resolved the discrepancies by revising CESSAR-DC Tables 3.2-1 and 3.2-2 and Figure 9.3.4-1. In addition, ABB-CE responded to GSI-23 by integrating into the CVCS a dedicated seal injection system (DSIS). The DSIS is considered a part of the CVCS and provides a diverse and reliable means of seal water injection for RCP seal cooling. Although the CVCS and DSIS are considered, and found acceptable by the staff, as non-safety-related systems, the RCP seal injection-related portions of the CVCS and DSIS as noted in CESSAR-DC Figure 9.3.4-1 (Amendment Q) are ASME Code Class 3, seismic Category I and Quality Group A. This classification assures enhanced reliability and availability of the RCP seal injection system and therefore, is acceptable. CESSAR-DC Tables 3.2-1 and 3.2-2, Amendment U, included DSIS components. On this basis, DSER Open Item 3.2.2-2 is resolved.

In response to RAI Q210.7 and DSER Open Item 3.2.2-3, which questioned ABB-CE's classification of the new- and spent fuel racks as non-nuclear-safety and with a QA requirement of N (not subject to 10 CFR Part 50, Appendix B requirements), ABB-CE reclassified the new and spent fuel racks as Safety Class 3, seismic Category I, and QA Class 1. The revised classification meets the applicable QA requirements of Appendix B to 10 CFR Part 50; is commensurate with the importance of the safety function of the new and spent fuel racks; and is, therefore, acceptable. On this basis, DSER Open Item 3.2.2-3 is resolved.

In response to RAI Q210.9, in which the staff questioned ABB-CE's commitment to specific portions of RG 1.151, "Instrument Sensing Lines," Revision 0, ABB-CE revised CESSAR-DC Sections 3.2.1 and 3.2.2 (Amendment N) by stating that the COL applicant will construct instrument-sensing lines and their supports per RG 1.151, in particular, positions relating to applicable seismic and ASME Code criteria. Also, ABB-CE revised CESSAR-DC

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Table 1.8-1 (Amendment J) to reference CESSAR-DC Sections 3.2.1 and 3.2.2. The staff concludes that RG 1.151 Positions C.2 and C.3, which provide guidance on the above issues, are satisfied. On this basis, DSER Confirmatory Item 3.2.2-1 is resolved.

In RAI Q210.10, the staff noted that the safety class and QA requirements for piping and component supports were not clearly defined in the CESSAR-DC. In response, ABB-CE modified CESSAR-DC Section 3.2.2 (Amendment K) to state that piping supports and component supports are of the same safety class and have the same QA requirements as the piping and components to which they apply. This CESSAR-DC statement is acceptable because the piping and component supports shall be classified in accordance with RG 1.26 to the same safety class of their associated piping and components. On this basis, DSER Confirmatory Item 3.2.2-2 is resolved.

On the basis of its review of CESSAR-DC Section 3.2.2 and applicable P&IDs, the staff concludes that the quality group classifications of all pressure-retaining and non-pressure-retaining SSCs important to safety that are identified in CESSAR-DC Table 3.2-1 are in conformance with RG 1.26, Revision 3, and with staff positions on previously licensed pressurized-water-reactor (PWR) plants and are, therefore, acceptable. Table 3.2-1, in part, identifies major components in fluid systems (such as pressure vessels, Hxs, storage tanks, pumps, piping, and valves) and in mechanical systems (such as cranes, refueling platforms, and other miscellaneous handling equipment). In addition, P&IDs in the CESSAR-DC identify the classification boundaries of interconnecting piping and valves. All of the above SSCs are constructed in conformance with applicable ASME Code and industry standards. Conformance to RG 1.26, previous staff positions, and applicable ASME Codes and industry standards provides assurance that component quality will be commensurate with the importance of the safety function of these systems. This constitutes the basis for complying with GDC 1 and is, therefore, acceptable.

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

All seismic Category I structures exposed to wind forces are designed to withstand the effects of the design wind specified in CESSAR-DC Table 2.0-1. Procedures used to

transform the wind velocity into pressure loadings on structures are in accordance with American Society of Civil Engineers (ASCE) 7 (Formerly ANSI A58.1-1982), ASCE Paper 3269, and ASCE Paper 4933. The plant design with respect to capability of the structures to withstand design wind loadings is acceptable and meets the requirements of GDC 2. The design reflects, as described in SRP Section 3.3.1,

- (1) appropriate consideration for the most severe wind not to exceed the velocities presented in CESSAR-DC Table 2.0-1 for future sites;
- (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and
- (3) the importance of the safety function to be performed.

These requirements are being met by the use of ASCE 7 and ASCE Papers 3269 and 4933, to transform the wind velocity into an effective pressure on structures and for selecting pressure coefficients corresponding to the structural geometry and physical configuration.

The plant structures are being designed with sufficient margin to prevent structural damage during the most severe wind loadings at a specific site not exceeding that specified in CESSAR-DC Table 2.0-1 so that the requirements of Item 1 (above) are met. In addition, in the design of seismic Category I structures, load combinations that occur as a result of the most severe wind loads and the loads resulting from normal and accident conditions have been taken into account (CESSAR-DC Section 3.8) as required by Item 2 (above).

The procedures used to determine the design wind loadings on structures for the plant have been used in the design of conventional structures and provide a conservative baseline that, together with other engineering design considerations, ensures that the structures will withstand such environmental forces. The use of these procedures gives reasonable assurance that, in the event of design-basis winds, the structural integrity of the applicable plant structures will not be impaired and, in consequence, safety-related systems and components located within these structures are adequately protected and will perform their intended safety functions, if needed, thus satisfying the requirements of Item 3 (above). Accordingly, the design is acceptable with respect to the determination of design wind loadings.

By DSER COL Action Item 3.3.1-1, the staff noted that the COL applicant must ensure that the velocity of the wind as presented in CESSAR-DC Table 2.0-1 will not be

exceeded by the site-specific design-basis wind. Also, any deviations from the plant arrangement of the System 80+ with respect to location or orientation of various buildings must be reviewed and accepted by the staff. ABB-CE incorporates the above information into CESSAR-DC Section 2.0. This is acceptable. This is included in COL Action Item 2.0-1.

3.3.2 Tornado Loadings

The staff position with regard to design-basis tornadoes (DBTs) is contained in two documents written in 1974: WASH-1300 (U.S. NRC, "Technical Basis for Interim Regional Tornado Criteria") and RG 1.76, "Design Basis Tornado for Nuclear Power Plants," Revision 0. According to WASH-1300, the probability of occurrence of a tornado that exceeds the DBT should be on the order of $10^{-7}/\text{year}$ for each nuclear power plant. The RG delineates the maximum wind speeds of 579 kilometers per hour (km/hr) (360 miles per hour (mph)) for the contiguous United States.

The staff reevaluated the regulatory positions in RG 1.76 using the considerable quantity of tornado data which has become available since the RG was developed. The reevaluation is discussed in NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," dated May 1986. The staff's interim position on RG 1.76 was issued in the March 25, 1988 letter, "ALWR Design Basis Tornado." In this interim position, the staff concluded that the maximum tornado wind speed of 531 km/hr (330 mph) is acceptable. In SECY-93-087, the staff recommended that the Commission approve its position that a maximum tornado wind speed of 483 km/hr (300 mph) is to be the DBT employed in the design of evolutionary and passive ALWRs. In its SRM dated July 21, 1993, the Commission approved the staff position.

ABB-CE states, in the CESSAR-DC, that all seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the plant are designed to resist tornado effects in accordance with the interim staff position in RG 1.76, and the tornado missile spectrum is in accordance with SRP Section 3.5.1.4. CESSAR-DC Table 2.0-1 specifies a maximum tornado wind speed of 531 km/hr (330 mph), maximum rotational tornado wind speed of 418 km/hr (260 mph), and a maximum translational tornado wind speed of 113 km/hr (70 mph). Also specified are a simultaneous atmospheric pressure drop to 16.5 kPa (2.4 psi) at the rate of 11.7 kPa/sec (1.7 psi/sec) and the radius of 45.7 m (150 ft). Because the values of these parameters are more severe than the tornado design-basis requirements specified in SECY-39-087 as approved in the July 21, 1993, SRM, the staff concludes that the ABB-CE System 80+ tornado design basis is acceptable.

The procedures used to transform the tornado wind velocity into pressure loadings are the same as for the winds discussed in Section 3.3.1 of this report. The tornado missile effects are determined using procedures discussed in CESSAR-DC Section 3.5. The tornado loadings are tornado wind pressure, internal pressure by tornado-created atmospheric pressure drop, and forces generated by the impact of tornado missiles. These loadings are combined with other loads as described in CESSAR-DC Section 3.8.

In view of the foregoing, the staff concludes that ABB-CE meets the requirements of GDC 2 and the guidelines of SRP Section 3.3.2 with respect to the structural capability to withstand design tornado wind loading and tornado missiles. ABB-CE meets the requirements of GDC 2 with respect to the capability of the structures to withstand design tornado wind loading and tornado missiles so that their design reflects

- (1) appropriate consideration of the most severe tornado characterized by the tornado parameters mentioned above
- (2) appropriate combinations of the effects of this severe natural phenomena with those resulting from normal plant operation and/or accident conditions
- (3) the importance of the safety function to be performed

These requirements are met by using interim RG 1.76 and the methods of transforming the tornado wind velocity into an effective pressure on structures as described in CESSAR-DC Section 3.3.1.

By committing design loads and load combinations to meet the guidelines of SRP Section 3.8, ABB-CE is designing the plant structures with sufficient margin to prevent structural damage during the most severe tornado loadings, so the requirements of Item 1 (above) are met. In addition, the design of seismic Category I structures, as required by Item 2 (above) includes, in an acceptable manner, load combinations of the most severe tornado load and loads resulting from normal plant operation and accident conditions. The procedures to determine the loadings on structures induced by the DBT specified for the plant have been used in the design of conventional structures for most severe winds and serve as a conservative baseline, which, together with other engineering design considerations, will ensure that the structures will withstand such severe environmental forces.

The use of these procedures gives reasonable assurance that, in the event of a DBT, the structural integrity of

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applicable plant structures will not be impaired and, consequently, safety-related systems and components located within these structures will be adequately protected and will perform their intended safety functions if needed, thus satisfying the requirement of Item 3 (above).

By DSER COL Action Item 3.3.2-1, the staff noted that the COL applicant must ensure that tornado parameters as described above and presented in CESSAR-DC Table 2.0-1 shall not be exceeded by those of the site-specific DBT. ABB-CE incorporates the above information into CESSAR-DC Section 2.0. This is acceptable. This is included in COL Action Item 2.0-1.

3.4 Water Level (Flood) Design

The System 80+ standard design requires flood protection, from both external and internal sources, for all SSCs whose failure could prevent safe shutdown of the plant or could result in an uncontrolled release of radioactivity. The staff reviewed the information in CESSAR-DC Section 3.4 on design of flood levels, flood protection design measures, and the information submitted by ABB-CE in response to the staff's RALs. The staff's review and acceptance criteria are in accordance with RG 1.59, "Design Basis Floods for Nuclear Power Plants," Revision 2, with regard to the methods utilized for establishing the probable maximum flood (PMF), probable maximum precipitation (PMP), seiche, and other pertinent hydrologic considerations. In addition, the staff's review and acceptance criteria are in accordance with RG 1.102, "Flood Protection for Nuclear Power Plants," Revision 1, as related to the means for protecting structures and systems important to safety from the effects of the PMF and PMP.

3.4.1 Flood Protection

CESSAR-DC Section 3.4 states that all seismic Category I structures, components, and equipment are designed for applicable loading caused by postulated floods. Specifically, the elevation level for floods at the reactor site is determined in accordance with RG 1.59 and ANSI/ANS 2.8, "Determining Design Basis Flooding at Power Reactor Sites." The finished yard grade adjacent to safety-related structures will be maintained at least 0.3 m (1 ft) below the ground floor elevation and no exterior access openings are lower than 0.3 m (1 ft) above plant grade elevation. Waterstops are used in all horizontal and vertical construction joints in all exterior walls up to flood level elevation. Walls subject to flooding are waterproofed and all penetrations in exterior walls up to the flood level elevation are sealed against the intrusion of water.

By DSER COL Action Item 3.4.1-1, the staff noted that the maximum site-specific flood levels and other safety-related structures where flood protection measures are required for the site will be addressed in the site-specific SAR. Subsequently, ABB-CE revised CESSAR-DC Section 2.4.1 to address this COL action item. This is included in COL Action Item 2.4-1. CESSAR-DC Section 2.4.1 states, in part, that the site-specific flooding projections will consider severe precipitation, snow melt, flooding due to ice cover, river flooding, ocean flooding, tsunami flooding, seiche effects, wave and storm surge effects, hurricane effects, high lake levels, and any other effects appropriate for the specific site. CESSAR-DC Section 3.4.2 also states that site-specific information will include a description of the site and elevation for all safety-related structures, exterior accesses, equipment, and systems. The staff finds that all the considerations for projecting maximum heights of site flooding events have been addressed in the CESSAR-DC. This is acceptable.

CESSAR-DC Section 3.4.4 states that seismic Category I structures are designed with flood protection measures in accordance with RG 1.102. Flood barriers are integrated into the design to provide additional flood protection while minimizing the impact on maintenance accessibility. Floods are controlled in the plant by the divisional structural walls which serve as a barrier between redundant divisions of safe shutdown systems and components, so that a single flooding event will not affect redundant safety systems. The lowest elevation of the structure wall have no door or other passage, and the limited penetrations through the lower wall are sealed against water. Where flood doors are provided, open and close sensors will indicate status. CESSAR-DC Section 3.4.4.1 states that a site-specific evaluation will be performed to ensure that all penetrations in seismic Category I structures below the external flood level are properly sealed to protect safety-related equipment. This is acceptable.

The primary means of flood control in the NA and RB subsphere is provided by the divisional wall which serves as a barrier between redundant trains of safe shutdown systems and components. Each half of the spherical containment subsphere is compartmentalized to separate safe shutdown components to the extent practical, while maintaining accessibility to the compartment. The subsphere, which houses the front-line safety systems is compartmentalized into four quadrants, with two quadrants on either side of the divisional structural wall. Although flood barriers separate the quadrants, the capability of removing equipment will be maintained. Emergency feedwater (EFW) pumps are located in separate compartments within the quadrants, and each compartment is protected by flood barriers. Penetrations are sealed and no doors are provided in the divisional wall that separates the

NA and RB subsphere up to elevation (EL.) 21.3 m (70 ft), which is the maximum internal flood. Safety-related electrical components are located at higher elevations such that flood events will not affect components. Flood barriers also give separation between electrical equipment and fluid mechanical systems at the lowest elevations within the NA. Curbs provide similar separation at higher elevations. CESSAR-DC Section 3.4.4.1 states that a site-specific flood analysis will ensure that all seismic Category I structures are protected against flood damage. This meets GDC 2 and is, therefore, acceptable.

The NA floor drainage systems are separated by division and Safety Class 3 valves will prevent backflow of water to areas containing safety-related equipment. Each subsphere quadrant is provided with redundant sump pumps (Safety Class 3) and associated instrumentation, which are powered from the diesel generators in the event of loss of offsite power.

Reducing the potential for possible flood sources is incorporated into the internal flood protection design. The CCW system and emergency feedwater systems (EFWS) are fully separated by division walls to prevent the migration of water between the divisions from a single flood source within one division. The station service water system (SSWS) is located outside the NA to eliminate a significant source of water. Safety Class 3 floor drain sump pumps and associated instrumentation in the diesel generator building will prevent flooding of the diesel generators. These pumps will be powered from the diesel generators in the event of loss of offsite power.

In response to RAI Q410.32(g) and RAI Q410.98, ABB-CE gave two different answers for the identification of seismic Category I structures requiring flood protection. Q410.32(g) asked ABB-CE to identify which of the structures listed in CESSAR-DC Table 3.2-4 will be designed using flood loading criteria. Q410.98 asked ABB-CE to discuss the internal flood protection methods to be used in the design as discussed in CESSAR-DC Section 3.4. Although the two responses were similar, the staff was not clear if the structures would be identified in CESSAR-DC Section 3.4.4.1 or by reference to CESSAR-DC Table 3.2-1. This was DSER Open Item 3.4.1-1.

In response to this concern, ABB-CE revised CESSAR-DC Section 3.4.4.1 which states, in part, that seismic Category I structures identified in CESSAR-DC Table 3.2-1 are designed for flood protection. The staff's review finds that Section 3.4.4.1 (flood protection measures) is consistent with Table 3.2-1 (classification of SSCs). Therefore, DSER Open Item 3.4.1-1 is resolved.

In addition, the seismic Category I diesel fuel oil storage structures, which ABB-CE added to CESSAR-DC Section 9.5.4 in response to RAI Q410.124(b), were not identified as structures requiring flood protection in accordance with RG 1.29. This was DSER Open Item 3.4.1-2.

In response to this open issue, ABB-CE revised CESSAR-DC Section 9.5.4 and added Section 9.5.4.1.2 to address the DFSS interface. Section 9.5.4.1.2 states, in part, that the seismic Category I portion of the diesel fuel oil storage structure is designed to withstand the effects of natural phenomena, including SSE, floods, tornados, and hurricanes. ABB-CE has added flood protection requirement to the design of DFSS. On this basis, DSER Open Item 3.4.1-2 is resolved.

On the basis of its review of the System 80+ design for flood protection, the staff finds that all of the DSER open issues as discussed above are resolved. Therefore, the staff concludes that the water level design for postulated floods is in accordance with SRP Section 3.4.1 and meets the requirements of GDC 2 and the guidance of RGs 1.59 and 1.102.

3.4.2 Analysis Procedures

CESSAR-DC Section 3.4.1 specifies that the design-basis flood level for the System 80+ plant is 0.3 m (1 ft) below plant finished yard grade. The maximum flood level for a specific site will be determined in accordance with RG 1.59 and ANSI/ANS 2.8-1983, "Determining Design Basis Flooding at Power Plant Sites." The staff requires, in accordance with CESSAR-DC Table 2.0-1, that the maximum flood level at a site be below the design-basis flood level. The design-basis flood is used in determining the applicable water level for the design of all seismic Category I structures, in accordance with the load combinations discussed in CESSAR-DC Section 3.8.4. The forces acting on those structures are determined on the basis of full external hydrostatic pressure corresponding to that flood level. All seismic Category I structures will be stable regarding both moment and uplift forces resulting from proper load combinations, including the design-basis flood level. These provisions will be incorporated into construction to maintain a dry environment in these structures during all flood conditions.

The analysis procedures for the design of the System 80+ plant are acceptable, as follows. ABB-CE has satisfied the review criteria of SRP Section 3.4.2 and the requirements of GDC 2, with respect to the structural capability to withstand the effects of the flood level and highest ground-water level. The design reflects the following, as described in SRP Section 3.4.2:

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- (1) appropriate consideration for the most severe flood not to exceed the flood level identified above for any future site
- (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena
- (3) the importance of the safety function to be performed

By limiting the design-basis flood elevation to a minimum of 0.3 m (1 ft) below the grade and the design groundwater level to 0.6 m (2 ft) below the grade, ABB-CE provides sufficient margin to prevent structural damage during the most severe flood described above, as well as the associated dynamic effects that are appropriate for the flood levels, so the requirement of Item 1 (above) is met. In addition, as required by Item 2 (above), the design of seismic Category I structures includes, in an acceptable manner, load combinations that will occur as a result of both the most severe flood or groundwater-related loads described above and loads resulting from normal and accident conditions.

The procedures used to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable. The procedures used to determine the external hydrostatic pressure corresponding to the flood level have been used in the design of conventional structures and prove to serve as a conservative baseline that ensures that structures will withstand such environmental forces.

The use of these procedures gives reasonable assurance that, in the event of floods as described above, the structural integrity of the plant seismic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions, thus satisfying the requirements of Item 3 (above).

By DSER COL Action Item 3.4.2-1, the staff noted that the COL applicant must provide a specific description of the site and the elevation for all safety-related structures, exterior accesses, equipment, and systems, from the standpoint of hydrology considerations and flood history (including date, level, peak discharge and related information for major historical flood events in the site region). The individual applicant will also address the following topics:

- PMP
- precipitation losses

- runoff and stream course models
- maximum flood flow
- water level determination
- coincident wind wave activity

CESSAR-DC Section 3.4.2 states that site-specific information will include a description of the site and elevation for all safety-related structures, exterior accesses, equipment, and systems. This is acceptable. This is included in COL Action Item 3.4-1.

By DSER COL Action Item 3.4.2-2, the staff noted that the COL applicant should ensure and demonstrate in the site-specific application that all seismic Category I structures are either protected against flood damage or are not subject to such damage. Hydrostatic and hydrodynamic effects of the flood must be considered and described for all postulated design flood levels for the conditions set for the future site, as outlined above. CESSAR-DC Section 3.4.2 states that all seismic Category I structures are protected against flood damage. This is acceptable. This is included in COL Action Item 3.4-1.

3.5 Missile Protection

The System 80+ standard design provides missile protection or redundancy for seismic Category I SSCs. The protection of essential SSCs is done by means of (1) minimizing the source of missiles, (2) orientation or physical separation of potential missile sources away from safety-related equipment and components, (3) missile containment through the use of protective shields and barriers near the source, and (4) hardening the safety-related equipment and components to withstand missile impact. The staff's review of the facility design for protecting SSCs important to safety against missiles is as follows.

3.5.1 Missile Selection and Description

CESSAR-DC Section 3.5.1 identifies potential missiles and characterizes them by type, source, and probability of occurrence, retention, and impact. For equipment with energy sources capable of creating a missile, the selection is based on the application of single-failure criterion to the retention features of the component. Where sufficient retention redundancy is provided in the event of a failure, no missile is postulated.

Internally-generated missiles can be generated from two types of equipment: rotating components and pressurized components. Rotating components include turbine wheels, fans, and auxiliary pumps and their associated motors.

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Pressurized components include valves, Hxs, vessels, and their associated components.

The types of missiles ABB-CE considered in the design of seismic Category I SSCs are discussed in the following sections of this report:

- Internally-generated missiles (Outside the Containment) - 3.5.1.1
- Internally-generated missiles (Inside the Containment) - 3.5.1.2
- Turbine Missiles - 3.5.1.3
- Missiles Generated by Natural Phenomena (Tornado) - 3.5.1.4
- Missiles Generated by Events Near the Site - 3.5.1.5
- Aircraft Hazards - 3.5.1.6

3.5.1.1 Internally-generated missiles (Outside the Containment)

The staff reviewed the design of the facility for protecting SSCs important to safety against internally-generated missiles outside the containment in accordance with SRP Section 3.5.1.1. Conformance to the acceptance criteria forms the basis for the staff's evaluation of the design for missile protection outside the containment with respect to the applicable regulations of 10 CFR Part 50. Specifically, GDC 4 (10 CFR Part 50, Appendix A) requires that these SSCs be protected against missiles. The review of internally-generated missiles does not include main turbine missiles which are evaluated in Section 3.5.1.3 of this report.

ABB-CE evaluated the potential missile sources from rotating and pressurized component failures. Precautions against missiles generated by rotating or pressurized equipment are made by reducing the potential for sources of missiles by equipment design, by orientation or physical separation of potential missile sources away from safety-related equipment, by missile containment through the use of protective shields near the source, and by hardening of those components that remain vulnerable. ABB-CE evaluated internal missile sources, such as auxiliary pumps and associated motors, valve stems and bonnets, and pressure vessels and concluded that none of these components are a credible source of missiles. Additionally, the CESSAR-DC states that the redundant safety systems outside the containment are physically separated so that no single gravitational or other type missile can impact both systems.

CESSAR-DC Section 3.5.1.1 states that there are no postulated missiles originating from auxiliary pumps and their associated motors outside containment for the following reasons:

- (1) The pump motors are induction type which have relatively slow running speeds and are not prone to overspeed. The motors are all pretested at full running speed by the motor vendor before installation.
- (2) The motor stator would tend to serve as a natural container of rotor missiles if there were to be any.
- (3) All pumps normally have relatively low suction pressures and, therefore, would not tend to be driven to overspeed due to a pipe break in the discharge line. Also, the induction motor would tend to act as a brake to prevent pump overspeed.
- (4) Industry experience shows that there have been no occurrences of impeller pieces penetrating pump casings.

ABB-CE states that no missiles are postulated from valves because all valve stems have a back-seat or shoulder that is larger than the valve bonnet opening. Motor-operated and manual valve stems are restrained by stem threads and operators on motor-, hydraulic-, and pneumatic-operated valves prevent ejection. Pneumatic-operated diaphragms and safety valve stems are restrained by spring force. Additionally, all valve bonnets are either pressure sealed, threaded or bolted such that there is redundancy for prevention of missile generation.

ABB-CE states that all pressurized vessels outside containment are moderate energy (1.90 MPa (275 psig)) or less and are designed and constructed to the standards of the ASME Code. In addition, all vessels will receive periodic in-service inspections and, where appropriate, will be provided with pressure relief devices.

On the basis of its review of the adequacy of the design for maintaining the capability for a safe shutdown in the event of internally-generated missiles outside the containment, the staff concludes that the above analysis is correct and that internally-generated missiles from rotating and pressurized components are not credible. The design is in conformance with the requirements of GDC 4 and meets the acceptance criteria of SRP Section 3.5.1.1 with respect to protection from internally-generated missiles outside the containment and is, therefore, acceptable.

3.5.1.2 Internally-generated missiles (Inside the Containment)

The staff reviewed the design of the facility for protecting SSCs important to safety against internally-generated missiles inside the containment in accordance with SRP Section 3.5.1.2. Conformance to the acceptance criteria

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forms the basis for the staff's evaluation of the design for missile protection inside the containment with respect to the applicable regulations of 10 CFR Part 50.

Protection for the SSCs inside the containment that are required for safe shutdown of the facility against postulated internally-generated missiles associated with plant operation, such as missiles generated by rotating or pressurized equipment, is provided by any one or a combination of barriers, separation, and equipment design. The primary means of protecting safety-related equipment from potential damage of the internally-generated missiles is shield walls and separation within the containment.

ABB-CE evaluated the potential missile sources inside the containment and states that the only credible missile sources identified are from the following high-energy systems:

- reactor vessel — control rod drive assembly, closure head nut and stud, and heated junction thermocouple assembly
- steam generator — studs and nuts of manways and handhole
- pressurizer — safety valve flange bolt, lower temperature element, and manway stud and nut
- RCP and piping — reactor coolant piping temperature nozzle with resistance temperature detector (RTD) assembly, surge and spray piping thermowells with RTD, and RCP thermowell with RTD

Other items considered and specifically excluded, because of redundant retention features, are valve stems, valve bonnets, and pressurized cover plates.

ABB-CE lists postulated missiles from equipment inside the containment in CESSAR-DC Table 3.5-1 and summarizes missile characteristics based on the criteria that either (1) the structures, shields, barriers, and equipment orientation protects the safety-related equipment from the primary missiles (above) and from any secondary missiles generated by their impact or (2) these missiles do not have sufficient energy to cause unacceptable impact or damage. Based on its review of CESSAR-DC Table 3.5-1, the staff finds ABB-CE's assumptions and evaluation for the internally-generated missiles inside the containment to be acceptable.

On the basis of its review of the adequacy of the facility design for protecting SSCs from internally-generated missiles inside the containment, the staff concludes that through the use of barriers, separation, and equipment

design, the facility design meets the requirements of GDC 4 and the acceptance criteria of SRP Section 3.5.1.2 and is, therefore, acceptable.

3.5.1.3 Turbine Missiles

The turbine generator placement and orientation for the System 80+ are illustrated in CESSAR-DC Figure 1.2-1. With respect to the RB, the turbine system is oriented so that a postulated turbine missile is not likely to strike the RB.

The probability of unacceptable damage from turbine missiles is expressed as the product of (1) the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure (IS)) fragments through the turbine casing (P_1); (2) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems or components (P_2); and (3) the probability of safety-related structures, systems, or components failing to perform their safety functions (P_3). CESSAR-DC Section 3.5.1.3 states that P_2 is less than 10^{-3} per turbine failure.

In previous reviews of the probability calculations, the staff found that the mathematical models used to calculate P_2 and P_3 required numerous approximations and simplifying assumptions in order to incorporate available data in the analysis. As a result, the calculations of P_2 and P_3 were not accurate; however, the calculation of P_1 is considered more accurate and precise. Therefore, in recent years, the staff placed emphasis on reviewing the probability of turbine missile generation (P_1). If P_1 is controlled to a minimum, the final failure probability of unacceptable damage can be met. Consistent with the staff position taken in recently licensed plants, the probability of turbine missile generation should be kept to no greater than 10^{-5} /reactor-year for an unfavorably oriented turbine and 10^{-4} for a favorably oriented turbine. As stated above, the turbine is favorably oriented. The staff also recommended certain actions for situations when the probability does not meet the minimum values required (see Table 3.1 of this chapter).

CESSAR-DC Section 3.5.1.3 states that the design follows the guidelines of RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," Revision 1. In a letter dated April 30, 1992 (LD-92-063), ABB-CE stated that the combined probability of turbine missile generation and adverse impact effects on seismic Category I systems and components is less than 10^{-4} /year.

By DSER COL Action Item 3.5.1.3-1, the staff noted that the COL applicant should submit a summary of the turbine maintenance and inspection program to ensure that the

Table 3.1 Criteria pertaining to the probability of turbine missile generation for a favorably oriented turbine

Probability criterion/year	Required licensee action
$P_1 < 10^{-4}$	This is the general, minimum reliability requirement for loading the turbine and bringing the system on line.
$10^{-4} < P_1 < 10^{-3}$	If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee is to take action to reduce P_1 to meet the first criterion before returning the turbine to service.
$10^{-3} < P_1 < 10^{-2}$	If this condition is reached during operation, the turbine is to be isolated from the steam supply within 60 days, at which time the licensee is to take action to reduce P_1 to meet the first criterion before returning the turbine to service.
$10^{-2} < P_1$	If this condition is reached at any time during operation, the turbine is to be isolated from the steam supply within 6 days, at which time the licensee is to take action to reduce P_1 to meet the first criterion before returning the turbine to service.

turbine missile generation probability (P_1) will be less than 10^{-4} /year. ABB-CE, in Amendment P, revised CESSAR-DC Section 3.5.1.3 to state that a summary of the turbine maintenance and inspection program including probability calculations of turbine missile generation will be submitted. This will ensure that the turbine missile generation probability be less than 10^{-4} /year for a favorably oriented turbine system (U.S. NRC, "Safety Evaluation Report Related to the Operation of Hope Creek Generating Station," NUREG-1048, Supplement No. 6, July 1986; and U.S. NRC, letter from C.E. Rossi to J.A. Martin, Westinghouse, Orlando, Florida, February 2, 1987) and is, therefore, acceptable. This is included in COL Action Item 3.5-1.

ABB-CE has sufficiently demonstrated to the staff, in accordance with RG 1.115, that the probability of turbine missile damage to SSCs important to safety is acceptably low. Therefore, the staff concludes that the turbine missile risk for the proposed plant design is acceptable and meets the requirements of GDC 4.

3.5.1.4 Missiles Generated by Natural Phenomena (Tornado)

The staff reviewed the tornado-generated missiles in accordance with SRP Section 3.5.1.4. Conformance to the acceptance criteria forms the basis for the staff's evaluation of the tornado-missile spectrum with respect to the applicable regulations of 10 CFR Part 50.

The staff did not use the portions of the SRP acceptance criteria concerning missile strike probability per year to damage safety-related systems. The review for this section of the SRP is concerned with establishing the missile spectrum, not with calculating the probability of damage.

The SRP acceptance criteria specify that the design should meet GDC 2 and 4. GDC 2 requires that SSCs important to safety be protected from the effects of natural phenomena, and GDC 4 requires that SSCs be protected against missiles. ABB-CE considered tornado-generated missiles as the only limiting hazard arising from natural phenomena that are of concern. ABB-CE selected Spectrum II per SRP Section 3.5.1.4 (Rev. 2) as the design-basis missile spectrum. The spectrum includes weight, impact velocity, and dimensions of the missile, and is in accordance with RG 1.76. The design impact velocities were selected for Region I, the region of highest velocity as defined in WASH-1300. On the basis of its review of the missile spectrum in CESSAR-DC Table 3.5-2, the staff finds that the spectrum is representative and acceptable at a site with a tornado velocity less than 531 km/hr (330 mph), a site parameter defined in CESSAR-DC Table 2.1.

The evaluation of the protection devices (barriers and structures) afforded safety-related equipment from the identified tornado missiles, including compliance with the guidelines of RG 1.117, "Tornado Design Classification," Revision 1, Positions C.1 through C.3, is addressed in Section 3.5.2 of this report. The adequacy of barriers and

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structures to withstand the effects of the identified tornado missiles is discussed in Section 3.5.3 of this report.

The tornado-missile spectrum is properly selected and meets the requirements of GDC 2 and 4, with respect to protection against natural phenomena and missiles, and the guidelines of RG 1.76, Positions C.1 and C.2, and RG 1.117, Positions C.1 through C.3, with respect to identification of missiles generated by natural phenomena. The tornado-missile spectrum also meets the acceptance criteria of SRP Section 3.5.1.4 and is, therefore, acceptable.

3.5.1.5 Missiles Generated by the Events Near the Site

By DSER COL Action Item 3.5.1.5-1, the staff noted that the missiles generated near the site will be addressed in the site-specific SAR. CESSAR-DC Section 3.5.1.5 states that justification will be provided in the site-specific SAR. This is acceptable. This is included in COL Action Item 3.5-1.

3.5.1.6 Aircraft Hazards

By DSER COL Action Item 3.5.1.6-1, the staff noted that the hazards due to aircraft impact near the site are considered to be site specific in the design of System 80+ and will be addressed in the site-specific SAR. CESSAR-DC Section 3.5.1.6 states that justification will be provided in the site-specific SAR. This is acceptable. This is included in COL Action Item 2.2-1.

3.5.2 SSCs To Be Protected From Externally Generated Missiles

The staff reviewed the design of the facility for protecting SSCs important to safety against externally-generated missiles in accordance with SRP Section 3.5.2. Conformance to the acceptance criteria forms the basis for the staff's evaluation of the design of the facility for providing protection from tornado-generated missiles with respect to the applicable regulations of 10 CFR Part 50.

The SRP acceptance criteria specify that the design should meet GDC 2 and 4. The spectrum of tornado missiles is discussed in Section 3.5.1.4 of this report. ABB-CE states that all safety-related systems, equipment, and components required to safely shut down the reactor and maintain it in a safe-shutdown condition are housed in seismic Category I structures designed as tornado-resistant structures.

In addition, protection from low-trajectory turbine missiles, including compliance with RG 1.115, "Protection Against

Low-Trajectory Missiles," Revision 1, is addressed in Section 3.5.1.3 of this report.

The System 80+ standard design regarding protection from externally-generated missiles is in accordance with the requirements of GDC 2 and 4 with respect to missile and environmental effects and the guidelines of RG 1.13 (Position C.2), RG 1.27 (Positions C.1 through C.3), RG 1.115, and RG 1.117 (Positions C.1 through C.3) with respect to protection of safety-related plant features from tornado missiles, including stored fuel and the ultimate heat sink. The design of the facility for providing protection from tornado-generated missiles also meets the applicable acceptance criteria of SRP Section 3.5.2 and is, therefore, acceptable.

3.5.3 Barrier Design Procedure

For the design of missile barriers, ABB-CE proposes several methods for estimating missile penetration depth for concrete structures. To ensure that no secondary missiles are generated as a result of spalling, a minimum barrier thickness of three times the estimated penetration formulas may be used if justified by full-scale impact tests. The staff's position is that, although no restriction is placed on the formulas to be used for local missile damage predication as long as it is technically sound, the minimum design wall and roof thicknesses documented in Table 1, "Minimum Acceptable Barrier Thickness Requirements For Local Damage Predication Against Tornado Generated Missiles," of SRP Section 3.5.3 must be complied with. This is because these concrete thicknesses are the minimum required for protection against Spectrum II tornado missiles proposed in the CESSAR-DC. By DSER Open Item 3.5.3-1, the staff noted that ABB-CE should incorporate Table 1 of SRP Section 3.5.3 into the CESSAR-DC. Subsequently, ABB-CE incorporated this information into CESSAR-DC Table 3.5-3 in Amendment N. This is acceptable and resolves DSER Open Item 3.5.3-1.

The local damage predication formula for steel structures in CESSAR-DC Section 3.5.3.1.2 is identical to one of the formulas specified in SRP Section 3.5.3, and is, therefore, acceptable.

The staff evaluated the procedures for the design of seismic Category I SSC to withstand the effects of missile impact in accordance with SRP Section 3.5.3. The evaluation included the review of the procedures for the prediction of local damage of concrete and steel structures and for the prediction of overall structural response due to missile impact as well as the adequacy of the missile parameters. The staff concludes that the barrier design procedure is in accordance with the guidelines of SRP

Section 3.5.3 and meets the requirements of GDC 2 and 4 and is, therefore, acceptable.

The procedures used by ABB-CE to determine the effects and loadings on seismic Category I structures and missile barriers induced by design-basis missiles selected for the plant are acceptable since these procedures are in compliance with the guidelines of SRP Section 3.5.3 and give a sound basis for engineering design to ensure that the structures or barriers are adequately resistant to and will withstand the effects of such forces. The use of these procedures provides reasonable assurance that in the event of design-basis missiles striking seismic Category I structures or missile barriers, the structural integrity of the structures, shields and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures are, therefore, adequately protected against the effects of missiles and will perform their intended safety functions, if needed. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of GDC 2 and 4.

3.5.4 General Design Bases

CESSAR-DC Section 3.5.4 requires that missile protection measures be provided for all seismic Category I SSCs. Specifically, appropriate missile barrier design procedures are used in designing systems and parts of systems located inside the containment to prevent a loss-of-coolant accident (LOCA) or preclude the systems' loss of their safety function from the impact of potential missiles. For systems and equipment outside containment, appropriate design features (e.g., proper turbine orientation, natural separation, or missile barriers) are used to ensure that any potential missile impact will not prevent the systems or equipment from performing their safety functions. Furthermore, the safety-related instrument and control systems and all the systems and equipment required to conduct a safe plant shutdown are protected from potential missile sources.

The general design requirements for missile protection have been adequately addressed in CESSAR-DC Section 3.5.4. The staff considers that the design bases conform to the guidelines provided in Section 3.5 of RG 1.70 and are, therefore, acceptable.

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

The staff reviewed the CESSAR-DC design for protecting SSCs against postulated piping failures outside containment in accordance with SRP Section 3.6.1. Conformance with the acceptance criteria forms the basis for the staff's evaluation of the CESSAR-DC design for providing protection against postulated piping failures outside the containment with respect to the applicable regulations of 10 CFR Part 50.

The SRP acceptance criteria specify that the design should meet GDC 4 as it relates to accommodating the dynamic effects of postulated pipe rupture, including the effect of pipe whipping and discharging fluids. The design is in compliance with GDC 4 if it conforms to SRP Section 3.6.1, BTP Auxiliary Systems Branch (ASB) 3-1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," in regard to high- energy and moderate-energy fluid systems outside the containment. BTP ASB 3-1 specifies that postulated piping failures in fluid systems should not cause a loss of function of essential safety-related systems and that nuclear plants should be able to withstand postulated failures of any fluid system piping outside the containment, taking into account the direct results of such failure and the further failure of any single active component with acceptable off-site consequences.

The fluid-system piping is designed in accordance with the high-energy and moderate-energy pipe criteria of Appendix B of BTP ASB 3-1 to meet the above requirements with respect to protection against pipe breaks outside the containment. These criteria concern failures in high-energy and moderate-energy fluid systems. The CESSAR-DC identifies the high-and-moderate-energy piping systems in accordance with the guidelines of SRP Section 3.6.1, BTP ASB 3-1, and also identifies those systems requiring protection from postulated piping failures.

The plant design accommodates the effects of postulated pipe breaks by means of physical separation, enclosure insuitably designed structures or compartments, drainage systems, pipe whip restraints, and equipment shields. Separation between redundant safety systems with their auxiliary supporting features is the basic protective measure.

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Based on its review of the criteria for establishing the effects of high-energy and moderate-energy pipe failure on safety-related systems and structures, the staff concludes that the protection provided against pipe failure outside the containment conforms with the guidelines of BTP ASB 3-1.

There are no high-energy lines in the vicinity of the control room. Therefore, there are no effects upon the habitability of the control room by pipe break, either from pipe whip, jet impingement, or transport of steam. Control room habitability systems are discussed further in Section 6.4 of this report.

The CESSAR-DC identifies the essential systems which are required to be protected from the effects of postulated piping failures. By means of design features, such as separation, barriers, and pipe whip and jet impingement restraints, the effects of pipe break will not damage essential systems to an extent that would impair their safety function. The potential environmental effects of steam on essential systems are discussed in Section 3.11 of this report.

On the basis of its review, the staff concludes that the System 80+ plant is adequately designed and protected against postulated piping failures outside the containment, including the combination of pipe failure and single active failure. The staff concludes that the facility design for providing protection against postulated piping failures complies with the requirements of GDC 4 and the guidance of BTP ASB 3-1 with respect to the protection of safety-related systems and components from postulated high- and moderate-energy line breaks. The design of the facility for providing protection against postulated piping failures also meets the acceptance criteria of SRP Section 3.6.1.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

GDC 4 requires that SSCs important to safety be designed to be compatible with and to accommodate the effects of the environmental conditions resulting from normal operations, maintenance, testing, and postulated accidents, including LOCA. It also requires that they be adequately protected against dynamic effects (including the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures and from events and conditions outside the nuclear power plant.

CESSAR-DC Section 3.6.2 describes the design bases for locating breaks and cracks in piping inside and outside containment, the procedure used to define the thrust at the

break location, the jet impingement loading criteria, and the dynamic response models and results. On the basis of its review of the information in CESSAR-DC Section 3.6.2, the staff finds that it conforms to the modified guidelines of SRP Section 3.6.2, Revision 2, including BTP MEB 3-1, as described in the following.

In DSER Section 3.6.2 the staff noted three general issues. In the first of these general issues, DSER Open Item 3.6.2-1, the staff stated that, contrary to the representation that CESSAR-DC Section 3.6.2 contains descriptions of results relating to the determination of break locations and dynamic effects associated with the postulated rupture of piping, no such descriptions were found in CESSAR-DC Section 3.6.2. ABB-CE was asked to delete this inference. By Amendment N, ABB-CE revised CESSAR-DC Section 3.6.2 to eliminate the statement that descriptions of these results were contained in Section 3.6.2. In addition, CESSAR-DC Section 3.6 states that the COL applicant will supply final designs of high and moderate energy fluid systems. On this basis, DSER Open Item 3.6.2-1 is resolved. The other two issues relate to the general methodology for determining the location of pipe ruptures and their dynamic effects.

Relative to the second general issue, the staff noted in the DSER that, by letter LD-92-056 of April 30, 1992, ABB-CE had transmitted a draft of Section 7 of the System 80+ DSDG to the NRC for review. Section 7 of the DSDG provided detailed requirements for the design of piping; heating, ventilation, and air conditioning (HVAC); and cable tray/conduit systems and supports. In particular, detailed requirements for pipe rupture were included in the DSDG. However, the staff noted in DSER Section 3.6.2 that evaluation of the DSDG could not be completed until the entire DSDG was in final form. Accordingly, by DSER Open Item 3.2-1, ABB-CE was asked to submit the final version of the DSDG for staff review. Subsequently, a complete version of the DSDG, dated January 15, 1993, was submitted for review. Although not part of CESSAR-DC, the DSDG has been subjected to a number of staff audits following which the information necessary for the staff's safety evaluation of CESSAR-DC was extracted from the DSDG, modified or clarified as necessary, and subsequently included in Appendix 3.9A to CESSAR-DC by Amendment P. Results of the staff's satisfactory review of this appendix are discussed in Section 3.12 of this report. On this basis, DSER Open Item 3.2-1 is resolved.

Relative to the third general issue, the staff noted that CESSAR-DC Sections 3.6.2.1.3 and 3.6.2.2 contain the statement that dynamic effects of pipe ruptures in the reactor coolant loop, surge line (SL), shutdown cooling (SC) line, safety injection (SI) line, and main steamline

(MSL) inside the containment were to be eliminated from the System 80+ plant design basis by LBB evaluations. CESSAR-DC Section 3.6.3 described the LBB evaluations to be performed to eliminate these dynamic effects. However, by RAIs Q210.13, Q210.27, and other requests, the staff noted that, pending the resolution of RAI Q252.03 concerning the LBB evaluation procedure, the criteria in Section 3.6.2 of CESSAR-DC were applicable to all high- and moderate-energy fluid system piping and asked ABB-CE to delete or revise various sections of CESSAR-DC to state this position. Subsequently, ABB-CE responded to these staff requests by asking the staff to reconsider its position. The staff reaffirmed its position in DSER Section 3.6.2, and, by DSER Open Item 3.6.2-2, asked ABB-CE to revise CESSAR-DC Section 3.6.2 to reflect the reaffirmed staff position.

By RAI Q252.03 the staff stated that (1) LBB analyses were to be based on specific plant data and (2) pre-approval of LBB based solely on methods and procedures was not acceptable. Requirements for LBB analyses were also provided. Subsequently, as a result of interactions (described in part in DSER Section 3.6.3) with the staff, ABB-CE proposed an approach which was acceptable to the staff. As documented in DSER Section 3.6.3, ABB-CE proposed to perform bounding LBB analyses based on the acceptance limits of NUREG-1061, Volume 3, to derive acceptable stress limits for the LBB piping which are to be satisfied in the final design. By DSER Open Item 3.6.3-1, ABB-CE was asked by the staff to submit these bounding LBB analyses for review and approval for design certification. As discussed in Section 3.6.3 of this report, the staff finds these LBB bounding analyses acceptable for the System 80+ design. On the basis of staff's approval of LBB method for System 80+ design, it is acceptable to eliminate the dynamic effects of the pipe ruptures of those lines which are eliminated by LBB and noted in CESSAR-DC Section 3.6.2.1.3. On this basis, DSER Open Item 3.6.2-2 is resolved.

Coincident with publication of the DSER, a fourth general issue arose concerning the consideration of OBE effects in pipe break postulation criteria. One of the guidelines in SRP Section 3.6.2, BTP MEB 3-1, states that the analyses for the maximum stresses, stress ranges, and usage factors to be used for the determination of postulated high-energy and moderate-energy pipe break and crack locations should be based on loads which include the OBE. In SECY-93-087, the staff recommended the elimination of the OBE in the design process on the basis that it would not result in a significant decrease in the overall plant safety margin. The detailed basis for the staff's recommendation is discussed in Section 3.1.1 of this report. For pipe break postulation without OBE, the staff specified

alternative criteria that differed from SRP Section 3.6.2. ABB-CE has incorporated these criteria into CESSAR-DC Section 3.6.2 for System 80+ design and, therefore, meets the Commission approved staff position, as described in Section 3.1.1 of this report.

3.6.2.1 Pipe Break Criteria for High-Energy Piping Systems

CESSAR-DC Section 3.6.2.1 provides information on the criteria used to define break and crack locations and configurations in high-energy and moderate-energy fluid systems both inside and outside containment. The staff reviewed this information in accordance with SRP Section 3.6.2, Revision 2. The review extended to the criteria for ASME Code, Section III, Class 1, 2, and 3, and seismic Category I non-ASME high- and moderate-energy system piping and to the criteria for piping in containment penetration areas and in other than containment penetration areas.

The staff finds that the criteria in CESSAR-DC Section 3.6.1.1 for the definition of high- and moderate-energy fluid system piping are consistent with the criteria in SRP Section 3.6.1, BTP ASB 3-1, and are used correctly in CESSAR-DC Section 3.6.2 for identifying the high-energy fluid systems inside and outside containment, which are listed in CESSAR-DC Tables 3.6-3 and 3.6-4, respectively. The staff's review of the criteria used by ABB-CE to define break locations and configurations are discussed below.

3.6.2.1.1 Pipe-Break Criteria Within the Containment Penetration Areas

SRP Section 3.6.2, including BTP MEB 3-1, Revision 2, states that breaks need not be postulated in portions of high-energy fluid system piping located between the CIVs inside and outside the containment that are designed to meet ASME Code, Section III, Article NE-1120 and additional guidelines specified in BTP MEB 3-1. These guidelines include an augmented ISI program that must be implemented for those portions of piping within the break exclusion zone.

By RAI Q210.14 and DSER Open Items 3.6.2-3 and 3.6.2-4, the staff asked ABB-CE to revise the criteria in CESSAR-DC Section 3.6.2.1.4.1 F (in Amendment I) for piping in containment penetration areas. The staff also requested ABB-CE to clarify the design and ISI requirements specified in CESSAR-DC Section 3.6.2.1.4.1 F, Item 3 (in Amendment I), for these break exclusion zones. Furthermore, ABB-CE was asked not to revise the break location criteria for ASME Code Section III, Class 1 high-energy and moderate-energy fluid system piping in areas

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other than in containment penetration areas which were specified in CESSAR-DC Section 3.6.2.1.4.1 A as proposed in response to RAI Q210.14. In response, by Amendment N, ABB-CE modified CESSAR-DC Sections 3.6.2.1.4.1.A, and 3.6.2.1.4.1.F (which has been changed to 3.6.2.1.4.1.G in Amendment N and thereafter) in accordance with BTP MEB 3-1, Revision 2, as requested. On this basis, DSER Open Items 3.6.2-3 and 3.6.2-4 are resolved.

3.6.2.1.2 Pipe-Break Criteria Outside the Containment Penetration Areas

In earlier amendments to CESSAR-DC, ABB-CE specified the pipe-break criteria for outside containment penetration areas, which conforms to SRP Section 3.6.2, Revision 1. In response to the preceding staff requests in RAI Q210.14 and DSER Open Items 3.6.2-3 and 3.6.2-4, ABB-CE, by the CESSAR-DC, Amendment N, adopted the staff position for postulation of pipe breaks for ASME Class 1 piping as delineated in SRP Section 3.6.2, BTP MEB 3-1 Revision 2. However, in Amendment P, ABB-CE reinstated the pipe break criteria for Class 1 piping outside the containment penetration areas as delineated in SRP Section 3.6.2, Revision 1. The staff's review of this reinstated position is as follows.

Before issuance of Revision 2 of BTP MEB 3-1 in June 1987, breaks were postulated at intermediate locations between terminal ends of a pipe run if the maximum stress range as calculated by Eq. (10) is greater than $2.4 S_m$ and if the maximum stress range as calculated by either Eqs. (12) or (13) is greater than $2.4 S_m$, where Eqs. (10), (12), and (13) and S_m are as defined in ASME Section III, Subsection NB 3653. This staff position was implemented in many plants operating today. In Revision 2 of BTP MEB 3-1, the same criteria are maintained for break exclusion in the containment penetration areas. However, for other areas, the criteria were revised to require that breaks be postulated at any intermediate locations when only Eq. (10) predicts stresses of greater than $2.4 S_m$. The use of Eqs. (12) and (13) was eliminated. This resulted in an inconsistency in the Revision 2 criteria in that they allow higher limits in the containment penetration areas than in other areas. The break exclusion areas should provide a margin greater than (or at least equal to) the margin for areas outside the break exclusion area. To determine the impact of this inconsistency, the staff audited several calculations for both boiling-water reactors (BWRs) and PWRs which compared the number of postulated pipe breaks resulting from the use of Revision 1 and 2 criteria. These analyses indicated that the Revision 2 criteria result in a significant increase in the number of postulated breaks, which may be counter-productive in terms of enhancing plant safety. Therefore, the staff has recom-

mended that SRP Section 3.6.2 be revised to reinstate that portion of the Revision 1 criteria related to allowing the use of Eqs. (12) and (13) for the postulation of intermediate pipe breaks in ASME Class 1 piping system. On the basis of the above evaluation, the staff concludes that it is acceptable to reinstate the Revision 1 criteria related to allowing the use of Eqs. (12) and (13) in CESSAR-DC Section 3.6.2.1.4 A (Amendment P).

3.6.2.2 Pipe-Break Analysis Methods

By RAI Q210.15 through RAI Q210.24, RAI Q210.28 and RAI Q210.29, and additional comments from the review of Amendment R to CESSAR-DC, the staff made a number of requests that ABB-CE revise CESSAR-DC Section 3.6.2 to commit to many of the details of the break and crack location and configuration criteria of BTP MEB 3-1, Revision 2, 1987. ABB-CE has responded to all of these requests by committing to revise CESSAR-DC Section 3.6.2 in accordance with the staff's comments and BTP MEB 3-1, Revision 2. Subsequently, ABB-CE incorporated the requested changes into Amendment U to CESSAR-DC. On this basis, DSER Confirmatory Item 3.6.2-1 is resolved.

SRP Section 3.6.2 also states that for the final design approval, the following information should be provided:

- sketches of applicable piping systems showing the location, size, and orientation of postulated pipe breaks and the location of pipe whip or restraints and jet impingement barriers
- a summary of the data developed to select postulated break locations (this should include calculated stress intensities, cumulative usage factors and stress ranges as delineated in SRP Section 3.6.2, BTP MEB 3-1)

By RAIs Q210.12, Q210.25, and Q210.26, and DSER Open Item 3.6.2-5, the staff requested that ABB-CE submit provide the sketches and data. ABB-CE responded to these requests for detailed information by stating that it was not required for certification, that it depended on plant-specific information, and that it would be subject to revision until the specific details of piping and other plant design are finalized. However, as an alternative approach to address the lack of design detail regarding pipe rupture, ABB-CE committed to (1) include design considerations and guidance for pipe routing and postulated pipe rupture analyses and (2) prepare a set of sample piping layouts and analyses, which will include a preliminary postulated pipe rupture evaluation of an economizer feedwater line. By DSER Open Item 3.6.2-5, the staff noted that until the above-referenced sample layouts and analyses, as well as the entire DSDG, are reviewed, the staff could not make

a safety determination on the acceptability of ABB-CE's alternative approach to the design detail regarding pipe rupture.

Subsequently, ABB-CE submitted the report, "Design and Analysis of System 80+ Distribution Systems" (DADS), Revision 1, dated January 1993, for staff review. Appendix L of this report summarized the results of a sample ASME Code, Section III Class 2/3 stress analysis which included a postulated pipe break analysis. The feedwater economizer line in the RB was used as the sample model. This report referenced the DSDG and a "sample piping analysis specification" (SPAS). ABB-CE also submitted a draft of this specification, "System 80+ Design Certification Sample Piping Analysis Specification," dated January 15, 1993, for staff review. The information in the DSDG, Section 7.1.8, and the SPAS, Section 7, both entitled "Postulated Pipe Breaks," are identical to the information in CESSAR-DC Section 3.6.2.

Relative to these ABB-CE submittals, the staff notes that

- (1) The DADS, DSDG, and SPAS are not part of CESSAR-DC.
- (2) The DADS Appendix L sample feedwater economizer line-break analysis was considered only as an illustrative example of an analysis by ABB-CE and was not reviewed since the piping analysis guidelines in CESSAR-DC Appendix 3.9A and the piping benchmark program (see Section 3.12.4.3 of this report) will ensure that the piping stress analysis for break postulation will be done satisfactorily by the COL applicant.
- (3) The DSDG, Section 7.1.8, and the SPAS, Section 7, provide only information already contained in CESSAR-DC Section 3.6.2. Similarly, no additional information is provided by CESSAR-DC Appendix 3.9 A, Section 1.8.2, which merely references CESSAR-DC Section 3.6.2. Consequently, none of the DSDG, SPAS, or CESSAR-DC Appendix 3.9A provides any pipe break postulation criteria supplementary to those in CESSAR-DC Section 3.6.2.

However, as noted in the following discussion, ABB-CE has developed details of the methodology for pipe whip analyses of the System 80+ design. These details are included in CESSAR-DC Appendix 3.6A. The staff reviewed this appendix and found it acceptable. Therefore, pipe whip analyses performed in accordance with the criteria and methods of Appendix 3.6A will be acceptable.

The staff concludes that proper implementation of (1) the piping design criteria in CESSAR-DC Appendix 3.9A together with the piping benchmark program and (2) the pipe whip analysis guidelines in CESSAR-DC 3.6A will assure that the piping stress analyses for break postulation and the pipe whip analyses are done satisfactorily. On this basis, DSER Open Item 3.6.2-5 is resolved.

CESSAR-DC Section 3.6.2.2 describes the analytical methods used to define the forcing functions and response models for the dynamic effects due to pipe breaks. Based on its review of the CESSAR-DC Section 3.6.2.2.1 methodology for the determination of pipe thrust and jet loads, the staff finds that the methods are in accordance with SRP Section 3.6.2, Revision 2, and, hence, acceptable. However, in part by RAI Q210.26, the staff asked ABB-CE to clarify the use of the ANSI/ANS-58.2 standard "Design Bases for Protection of Light-Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," for calculating fluid thrust forces. In response to this request, ABB-CE satisfactorily demonstrated how the methodology of Appendix B of the standard was used to develop the one-step forcing function described in CESSAR-DC Section 3.6.2. These forcing functions were shown to be at least as conservative as the forcing functions described in the standard. DSER Section 3.6.2 stated erroneously that, by RAI Q210.26, the staff had requested ABB-CE to cite the 1988 Edition of the ANSI/ANS 58.2 standard. The DSER also stated that ABB-CE had committed to comply with this request (DSER Confirmatory Item 3.6.2-2). However, the edition of the standard cited by CESSAR-DC was the 1988 Edition. Accordingly, DSER Confirmatory Item 3.6.2-2 is withdrawn and is resolved.

The staff reviewed methods for dynamic analysis of pipe whip described in CESSAR-DC Section 3.6.2.2.2. As part of its assessment of ABB-CE's methods of dynamic analysis of pipe whip, by RAI Q210.30, the staff asked ABB-CE to submit justification for the elastic-plastic moment curvature law used in the finite difference pipe whip analysis method described in the CESSAR-DC Appendix 3.6A. ABB-CE responded to RAI Q210.30 by submitting a general description of analysis methods only and deleting CESSAR-DC Appendix 3.6A in its entirety. This response did not give sufficient information to assess the acceptability of the methods of dynamic analysis of pipe whip for the System 80+ plant design. Subsequently, in response to requests made in the staff audits that ABB-CE submitted details of the methodology for pipe whip analyses of the System 80+ design, ABB-CE, by Amendment P, included in Appendix 3.6A descriptions of the energy balance and lumped parameter analysis methods. Furthermore, as previously noted by staff, neither the DSDG Section 7.1.8 nor CESSAR-DC Appendix 3.9 A,

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Section 1.8.2, provided pipe break information supplemental to that in CESSAR-DC Section 3.6.2.

By Amendments T and U to CESSAR-DC, ABB-CE revised Appendix 3.6A to submit guidelines for the design and selection of pipe-whip restraints, and pipe rupture and jet impingement evaluation methods. The pipe-rupture evaluation methods include energy balance and dynamic analysis for simplified and detailed models. The staff has reviewed Appendix 3.6A and found it to be in accordance with the guidelines of SRP Section 3.6.2, Revision 2, and hence acceptable. On this basis, DSER Open Item 3.6.2-6 is resolved.

Additionally, the staff conducted audit reviews to further assess ABB-CE's methods of dynamic analysis of pipe whip against the BTP MEB 3-1 B.1.c.(4) criterion that structures separating high-energy lines from safety-related components be designed to withstand the consequences of the pipe break in the high-energy line. This produces the greatest effect at the structure irrespective of the fact that the break postulation stress criteria might not require such a break location be postulated.

CESSAR-DC Section 3.6.2.1.1 commits to comply with the BTP MEB 3-1 criterion except that structures inside the containment are designed to withstand the dynamic load effects of postulated pipe breaks not eliminated by the LBB method. This is acceptable on the basis of staff's approval of LBB method for System 80+ design as discussed in Section 3.6.3 of this report. As such, the dynamic effects of pipe breaks inside the containment need not be considered for piping systems evaluated by LBB.

CESSAR-DC Section 3.6.1.1 provides design bases for postulated piping failures in fluid systems and indicates that protection of vital equipment is achieved by separation of redundant safe shutdown systems and by separation of high energy piping systems from each safe shutdown system. More specifically, safe shutdown systems inside the containment are protected by separation and barriers, including the secondary shield wall, refueling cavity walls, and the operating floors. Most safe shutdown systems outside containment are located in the RB subsphere and NA, and divided by a structural wall which serves as a barrier between redundant trains of safe shutdown systems. The main steam, steam generator blowdown, and main feedwater lines outside containment are separated from safe shutdown systems by locating the essential portions of these systems in the main steam valve houses (MSVHs). The high-energy portions of the CVCS lines in the NA are 5-cm (2-in.) lines which are located in the vicinity of the 122-cm (4-ft) thick NA divisional wall.

During the audits, the staff asked ABB-CE to provide for staff review sample calculations for the dynamic effects of worst-case breaks in the MSL in the MSVH, and in the CVCS line in the vicinity of the NA divisional wall. The staff found that the dynamic effects on the MSVH due to the worst-case MSL breaks was conservatively evaluated by ignoring the restraint of MSL whip restraints. The effects of the worst-case CVCS line break were also evaluated in accordance with the procedures in CESSAR-DC Appendix 3.6A, which was reviewed and found acceptable by the staff as discussed above. On the basis of the results of these audit reviews, the staff concludes that ABB-CE's evaluation of the dynamic effects of pipe breaks is in accordance with BTP MEB 3-1 criteria and, hence, acceptable.

CESSAR-DC Section 3.6.2.3 describes the dynamic analysis methods used to verify the design adequacy of SSCs supports under pipe whip or jet impingement loadings.

By RAIs Q210.31 and Q210.32, the staff asked ABB-CE to clarify the allowable stresses and dynamic load factors used in the pipe whip analyses as described in the CESSAR-DC Section 3.6.2.3. ABB-CE responded to these requests by committing to using a 50-percent ultimate strain criterion and a dynamic load factor of 2 for other than energy-absorbing whip restraint structural elements. These criteria are in accordance with SRP Section 3.6.2, Revision 2, and, hence, acceptable.

3.6.2.3 Conclusions

On the basis of its review of Section 3.6.2 of the CESSAR-DC, the staff concludes that the criteria for postulating pipe rupture and crack locations not eliminated by LBB and the methodology for evaluating the subsequent dynamic effects resulting from these ruptures comply with SRP Section 3.6.2, meet GDC 4 and, therefore, are acceptable for ensuring that the System 80+ plant design is adequately protected against the effects of postulated high-energy line breaks. The staff's conclusion is based on the following.

The pipe rupture locations will be determined using the staff-approved criteria and guidelines just discussed. The design methods for mitigation devices for high-energy pipe rupture and the measures to deal with the subsequent dynamic effects of pipe whip and jet impingement have been sufficiently and adequately defined by ABB-CE to provide adequate assurance that upon completion of the high-energy line break analyses by the COL applicant, the ability of safety-related SSCs to perform their safety functions will not be impaired by the postulated pipe ruptures.

The provisions for protection against the dynamic effects associated with pipe ruptures of the RCPB inside the containment and the resulting discharging fluid provides adequate assurance that design-basis LOCAs will not be aggravated by the sequential failures of safety-related piping and that the performance of the emergency core cooling system will not be degraded as a result of these dynamic effects.

The final arrangement of piping and restraints and the final design considerations for high-energy and moderate-energy fluid systems inside and outside the containment, including the RCPB, shall be the responsibility of the COL applicant to complete and shall use the staff-approved high-energy line break criteria and guidelines discussed above to ensure that the SSCs important to safety that are in close proximity to the postulated pipe ruptures will be protected. CESSAR-DC Section 3.6 states that the COL applicant will provide final designs of high-energy and moderate-energy fluid systems. This is acceptable. This is included in COL Action Item 3.6-1. In using the criteria and guidelines discussed above, the staff is assured that the consequences of pipe ruptures not eliminated by LBB will be adequately mitigated so that the reactor can be safely shut down and can be maintained in a safe-shutdown condition in the event of a postulated rupture of a high-energy or moderate-energy piping system inside and outside the containment.

3.6.3 LBB Evaluation Procedure

In CESSAR-DC, Amendment E, Section 3.6.3.9, ABB-CE stated that the Class 1 piping with a diameter of 25-cm (10 in.) or greater and the MSL piping meet all the criteria for the application of LBB. The LBB analysis is used to eliminate the dynamic effects of postulated pipe ruptures.

Under the broad-scope revision to GDC 4 (52 FR 41288; October 27, 1987), the NRC allows the use of advanced technology to exclude from structural design consideration the dynamic effects of pipe ruptures in nuclear power plants, provided it is demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design bases for the piping. The demonstration of low probability of pipe rupture utilizes a deterministic fracture mechanics analysis that evaluates the stability of postulated, small, through-wall flaws in piping and the ability to detect leakage through the flaws long before the flaw could grow to unstable sizes and rupture the pipe. The concept underlying such analyses is referred to as LBB.

3.6.3.1 LBB Acceptance Criteria

GDC 4 states, in part, that "dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

The analyses referred to in GDC 4 should be based on such specific data as piping geometry, materials, and piping loads. The staff must review the LBB analyses for specific piping design before ABB-CE can exclude the dynamic effects from the design basis. Applicants seeking design certification for ALWRs under 10 CFR Part 52 should establish preliminary stress analysis results, provided bounding limits (both upper and lower bound) are determined, in order to establish assurance that adequate margins are available for leakage, loads, and flaw sizes. These bounding values and preliminary analyses should be verified when as-built and as-procured information becomes available during the COL phase. Verification of the preliminary LBB analysis should be completed at the COL stage based on actual material properties and final, as-built piping analyses as part of inspection, test, analyses, and acceptance criteria (ITAAC) associated with 10 CFR Part 52 before fuel loading. The staff position on LBB application is stated in SECY-93-087 and the Commission approved it in its SRM dated July 21, 1993.

A margin of 10 on leakage is required so that leakage from the postulated flaw size is assured of detection when the pipe is subjected to normal operational loads. A margin of $\sqrt{2}$ on loads is required to ensure that leakage-size flaws are stable at normal plus accident loads (e.g., SSE and safety-relief valve discharge loads). A factor of 2 between the leakage-size flaw and the critical-size flaw is required to ensure an adequate stability margin for the leakage-size flaw. The analysis must be done for an entire pipe run from anchor to anchor.

In addition, applicants seeking approval of LBB during the design certification phase for an ALWR will be required to perform LBB analyses to establish through-wall flaw sizes and flaw stability. For through-wall flaw sizes, a lower-bound, normal-operational stress limit must be established for dead weight, pressure, and thermal loadings. The mean or best-estimate stress-strain curve should be used. For flaw stability, an upper-bound stress limit should be established for normal loadings plus SSE. A lower-bound stress-strain curve for base metal should be used regardless of whether the weld or base metal is limiting. In addition, a lower-bound toughness (weld metal or base metal) should be used.

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A deterministic fracture mechanics evaluation accounting for material toughness is required. Applicants may propose any fracture mechanics evaluation method for NRC staff review. However, ABB-CE will have to validate the method by comparing it with other acceptable methods or with experimental data.

3.6.3.2 LBB Limitations

The staff has established certain limitations on the use of the LBB approach for excluding piping that is likely to be susceptible to failure from various degradation mechanisms during service. A significant portion of the LBB review involves the evaluation of the susceptibility of the candidate piping in various degradation mechanisms to demonstrate that the candidate piping is not susceptible to failure from these degradation mechanisms. The NRC staff reviews the operating history and measures to prevent or mitigate these mechanisms.

The LBB approach cannot be applied to piping that can fail in service from such effects as water hammer, creep, erosion, corrosion, erosion-corrosion, fatigue, thermal stratification, and environmental conditions. Such piping is excluded because these degradation mechanisms challenge the assumptions in the LBB acceptance criteria. For example, (1) water hammer may introduce excessive dynamic loads that are not accounted for in the LBB analyses and (2) corrosion and fatigue may introduce flaws whose geometry may not be bounded by the postulated through-wall flaw in the LBB analyses. Adhering to the defense-in-depth principle, piping susceptible to failure from these potential degradation mechanisms is excluded from LBB applications.

The limitations and acceptance criteria for LBB used by the NRC staff are discussed in NUREG-1061, Volume 3, "Evaluation of Potential for Pipe Breaks, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," dated November 1984.

3.6.3.3 LBB Design Basis

The broad-scope rule introduced an acknowledged inconsistency in the design basis by excluding the dynamic effects of postulated pipe ruptures while retaining non-mechanistic pipe rupture for containments, emergency core cooling systems (ECCS), and environmental qualification (EQ) of safety-related electrical and mechanical equipment.

The NRC staff subsequently clarified its intended treatment of the containment, ECCS, and EQ in the context of LBB application in a request for public comments on this issue that was published on April 6, 1988 (53 FR 11311). Effects resulting from postulated pipe breaks can be

generally divided into local dynamic effects and global effects. Local dynamic effects of a pipe break are uniquely associated with that of a particular pipe break. These specific effects are not caused by any other source or even by a postulated pipe break at a different location. Examples of local dynamic effects are pipe whip, jet impingement, missiles, local pressurization, pipe break reaction forces, and decompression waves in the intact portions of that piping or communicating piping. Global effects of a pipe break need not be associated with a particular pipe break. Global effects can be caused by failures from such sources as pump seals, leaking valve packing, flanged connections, bellows, manways, rupture disks, and ruptures of other piping. Examples of global effects are gross pressurization, temperatures, humidity, flooding, loss of fluid inventory, radiation, and chemical condition.

The application of LBB technology eliminates the local dynamic effects of postulated pipe breaks from the design basis. However, global effects may still be caused by something other than the postulated pipe break. Since the global effects from the postulated pipe break provide a reasonably conservative design envelope, the NRC staff will continue to require the consideration of global effects for various aspects of the plant design such as EQ, ECCS, and the containment.

The elimination of local dynamic effects of postulated high-energy pipe breaks from the design basis of ALWRs using advanced fracture mechanics analyses (LBB approach) is permitted in the revised GDC 4 of 10 CFR Part 50, Appendix A.

3.6.3.4 ABB-CE LBB Evaluation Approach

The application of the LBB approach to ALWRs seeking design certification under 10 CFR Part 52 is acceptable when appropriate bounding limits are established during the design certification phase using preliminary analyses results and verified during the COL phase by the COL applicant performing the appropriate ITAAC discussed herein. Bounding analyses as described above must be done by ABB-CE requesting the application of LBB during the design certification phase. NUREG-1061, Volume 3, constitutes the current NRC-approved LBB methodology and acceptance criteria.

By letter dated May 8, 1992 (LD-92-064), ABB-CE stated that it will perform bounding LBB evaluations based on preliminary pipe design analyses for piping evaluated for LBB using the guidelines of NUREG-1061, Volume 3. For each piping system evaluated for LBB, potential degradation mechanisms, steam hammer and water hammer, and thermal stratification were to be considered, as applicable. In addition, dynamic strain aging of carbon

steel, environmental effects on fatigue, and thermal aging of cast stainless steel piping were to be considered in each LBB evaluation as appropriate. Each LBB piping system was to be evaluated from anchor point to anchor point. Leakage detection outside the containment would be considered for the MSL if the anchor-to-anchor portion of the piping evaluated includes piping that can leak outside the containment. A leak detection capability of 0.06 L/sec (1.0 gpm) was to be used with a factor of 10 for calculating the length of a leakage crack. ABB-CE also submitted a reference for benchmarking its LBB calculations.

In a meeting with the staff on April 22 and 23, 1992 (meeting summary dated May 28, 1992), ABB-CE proposed not to perform the actual routing of candidate piping for LBB analyses. ABB-CE proposed to use the LBB acceptance limits given in NUREG-1061, Volume 3, to derive corresponding acceptable stress limits for the LBB piping. ABB-CE stated that the acceptable stress limits would form a "window" in stresses and the COL applicant would have to verify that the piping is within this window during the COL stage to justify the application of LBB. This window concept is a new approach in meeting the staff's established LBB criteria. The staff noted that ABB-CE should perform the bounding LBB analyses based on the actual pipe routing. Since ABB-CE does not intend to complete the actual pipe routing, the staff cannot rule out the possibility that the COL applicant will not be able to satisfy this theoretical window. If the COL applicant cannot meet the limits in the window, appropriate hardware would have to be added for protection against a postulated pipe break. The hardware modification would be feasible except for the building structure that may already have been constructed. Thus, the staff noted that if the bounding LBB analyses are not based on actual pipe routing, the building subcompartments should be designed to accommodate a postulated pipe break of the LBB candidate piping.

Subsequently, the staff found that preliminary pipe routings were used in the bounding LBB analyses, and that stresses in LBB candidate piping systems were within the acceptance limits (or window) based on NUREG-1061, Volume 3. The preliminary routings were of conventional design. The stresses met the ASME Code and the LBB acceptance limits with ample margins. Consequently, the earlier staff concern was alleviated because it is extremely unlikely that the actual routings of the LBB candidate piping systems in the COL stage will not be able to meet the acceptance window limits. The staff also found that the related subcompartments are designed to withstand the pressures resulting from pipe breaks not eliminated by LBB, leakage cracks in piping for which LBB is approved, and postulated high-energy leakages from other sources. Thus the staff concluded that this ABB-CE approach was acceptable

and, by DSER Open Item 3.6.3-1, asked ABB-CE to submit its bounding LBB analyses for staff review and approval for design certification.

As part of the LBB bounding analysis review, the staff noted in the DSER that it would also evaluate ABB-CE's justification for using a conversion factor of 2.4 L/sec/cm² (250 gpm/in.²) of leak area for estimating leakage and ABB-CE's justification for combining loads for the flaw stability analysis. By letter dated May 8, 1992 (LD-92-064), ABB-CE further clarified these two subjects. Previously, for estimating leakage, the staff accepted the EPRI PICEP computer program referenced in CESSAR-DC Section 3.6.3.3.2 for other LBB applications. In DSER Section 3.6.3, the staff also stated that (1) the staff was concerned that ABB-CE is not applying state-of-the-art technology in estimating leakage and (2) for stability analysis, the staff was concerned that ABB-CE was not combining the appropriate transient loads for stability analysis. Subsequently, in Amendment R to CESSAR-DC, ABB-CE committed in Appendix 3.9A, Section 1.9.6.2, to use the PICEP program to estimate leakage. The staff review of the bounding analyses of the LBB candidate piping systems as documented in Section 3.6.3.5.3 of this report also found that loads were properly combined in these analyses. Accordingly, the earlier staff concern on appropriateness of combining the transient loads was resolved.

By DSER COL Action Item 3.6.3-1, the staff noted that the COL applicant should be required to verify that the actual material properties and final, as-built piping analyses meet the acceptance parameters established in the bounding LBB analyses. CESSAR-DC Section 3.6.3.8 incorporates the above information. This is acceptable. This is included in COL Action Item 3.6-1.

3.6.3.5 Review of ABB-CE Bounding LBB Analyses

Using the criteria of NUREG-1061, Volume 3, the staff reviewed the bounding LBB analyses submitted or presented during audit reviews and meetings by ABB-CE for compliance with GDC 4. ABB-CE submitted or presented these analyses in response to DSER Open Item 3.6.3-1. The results of this review are provided in the following:

For clarification, the staff notes that criteria and analyses in the DSDG, Section 7.1.9 and Appendix 7A; the SPAS, Section 9; and the DADS, Appendices A through J, relating to LBB analyses were not considered as part of CESSAR-DC. These documents are noted in Section 3.6.2.3 of this report. Instead, the staff conducted its reviews on the basis of the information in CESSAR-DC Section 3.6.3 and Appendix 3.9A, Section 1.9, and specific analyses found in the sections that follow.

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3.6.3.5.1 Piping Analyzed for LBB

CESSAR-DC Sections 3.6.2.1.3 and 3.6.3 indicate that LBB analyses are to be performed for the following piping systems:

- main coolant loop (MCL) piping, hot and cold legs
- surge line (SL)
- direct vessel injection (DVI) line (main run inside the containment)
- shutdown cooling (SC) line (main run inside the containment)
- MSL (main run inside the containment)

Design parameters used in the LBB analyses are listed in Table 3.2 of this report. The capability of the leak detection systems for each of the piping systems in Table 3.2 is 3.8 L/min (1.0 gpm). Except for the MSL, LBB has been applied to all of these types of systems in the System 80 and other PWR plants.

3.6.3.5.2 Acceptance Criteria for System 80+ LBB Analyses

For consistency with previous reviews, the staff used the following acceptance criteria for evaluating compliance of the ABB-CE System 80+ LBB analyses with GDC 4:

- (1) The loading conditions should include the static forces and moments (pressure, deadweight, and thermal expansion) due to normal operation and the forces and moments associated with the SSE. These forces and moments should be located where the highest stresses, coincident with the limiting material properties, are induced for base materials, weldments, and safe ends.
- (2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion cracking, fatigue, or water hammer are not likely, should be provided. Relevant operating history should be cited, which includes system operational procedures; system or component modification; water chemistry parameters, limits, and controls; and resistance of material to various forms of stress corrosion and performance under cyclic loadings.
- (3) The materials data provided should include types of materials and materials specifications used for base metal, weldments, and safe ends; the materials properties including the fracture mechanics parameter "J-integral" (J) resistance (J-R) curve used in

the analyses; and long-term effects such as thermal aging and other limitations to valid data (e.g., J maximum and maximum crack growth).

- (4) A through-wall flaw should be postulated at the highest stressed locations determined from criterion (1) above. The size of the flaw should be large enough so that the leakage is assured of detection with at least a factor of 10 using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.
- (5) It should be demonstrated that the postulated leakage flaw is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. The margin, in terms of applied loads, should be at least 1.4 and should be determined by flaw stability analysis, that is, the leakage-size flaw will not experience unstable crack growth even if larger loads (larger than design loads) are applied. However, the final rule permits a reduction of the margin of 1.4 to 1.0 if the individual normal and seismic (pressure, deadweight, thermal expansion, SSE, and SAM) loads are summed absolutely. This analysis should demonstrate that crack growth is stable and the final flaw size is limited, so that a double-ended pipe break will not occur.
- (6) The flaw size should be determined by comparing the leakage-size flaw to the critical-size flaw. Under normal plus SSE loads, it should be demonstrated that there is a margin of at least 2 between the leakage-size flaw and the critical-size flaw to account for the uncertainties inherent in the analyses and leakage detection capability. A limit-load analysis may suffice for this purpose; however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferable.

3.6.3.5.3 Results of Review of ABB-CE Bounding LBB Analyses

The ABB-CE approach for the design of System 80+ piping systems evaluated for LBB is described in CESSAR-DC Appendix 3.9A Sections 1.9.1 through 1.9.3. In this approach, LBB acceptance criteria are established for each of these systems. These criteria are provided in piping evaluation diagrams (PEDs) which define acceptable ranges of normal operating and "maximum design" loads (see below) for the systems. The PEDs are developed on the basis of current regulatory leakage and flaw stability requirements and on the system design parameters, including pipe sizes and materials and welding processes specified in Table 3.2 of this report.

Table 3.2 LBB analysis parameters

Piping system	Anchor to anchor description	Nominal pipe size	Material base metal	Weld type
MCL Hot leg	SG nozzle & RV nozzle	1070 mm (42") ID 89 mm (3 1/2") wall thickness	SA-516 Gr70 or SA-508 CL1A	SMAW
MCL Cold leg	SG nozzle & RV nozzle	762 mm (30") ID 64 mm (2-1/2") and 76 mm (3") wall thickness	SA-516 GR70 or SA-508 CL1A	SMAW
Surge line	HL nozzle & PZR nozzle	305 mm (12") Schedule 160	SA-312 TP316 or TP347	GTAW (tig)
Direct vessel injection	RV nozzle SI tank nozzle & con- tainment penetration	305 mm (12") Schedule 160	SA-312 TP304 or TP316	GTAW (tig)
Shutdown cool- ing	HL nozzle & con- tainment penetration	406 mm (16") Schedule 140	SA-312 TP304 or TP316	GTAW (tig)
Main steam	SG nozzle & con- tainment penetration	711 mm (28") ID 38 mm (1-1/2") wall thickness	SA-516 Gr70	SMAW

In principle, layouts of the systems in accordance with the design parameters of Table 3.2 which satisfy the acceptance criteria in the PEDs will also satisfy the regulatory leakage and flaw stability requirements. Consistent with DSER COL Action Item 3.6.3-1, CESSAR-DC Appendix 3.9A, Section 1.9.1, states that site-specific evaluation will confirm that the bases of the acceptance criteria are satisfied by the final as-built piping design and materials. This is acceptable.

The staff reviewed the following ABB-CE calculations for the development of the PEDs for the Table 3.2 piping systems:

- ABB-CE Calculation No. ALWR-ME-C-010, "LBB Piping Evaluation Diagram For a 12" Sch. 160 Pipe Such as the Surge Line For System 80+," March 16, 1993.
- ABB-CE Calculation No. ALWR-ME-C-011, "LBB Piping Evaluation Diagram For a 12" Sch. 160 Pipe Such as the Direct Vessel Injection Line For System 80+," March 16, 1993.
- ABB-CE Calculation No. ALWR-ME-C-012, "LBB Piping Evaluation Diagram For a 16" Sch., 140 Pipe Such as the Shutdown Cooling Line For System 80+," March 17, 1993.
- ABB-CE Calculation No. ALWR-ME-C-013, "LBB Piping Evaluation Diagram For a 28" ID 1 1/2" Thick Wall Pipe Such as the Main Steamline For System 80+," March 15, 1993.
- ABB-CE Calculation No. ALWR-ME-C-014, "LBB Piping Evaluation Diagram For a 42" ID, 3 1/2" Thick Wall Pipe, Such as the Main Coolant Hot Leg Line," April 2, 1993.
- ABB-CE Calculation No. ALWR-ME-C-015, "LBB Piping Evaluation Diagram For a 30" ID, 2 1/2" Thick Wall Pipe Such as the Main Coolant Cold Leg Line," April 1, 1993.

As a result of the discussion with the staff, ABB-CE incorporates the PEDs into CESSAR-DC Figures 3.9A-30 through 3.9A-36 of Appendix 3.9A. The staff also reviewed ABB-CE and Duke Engineering and Services, Inc. (DESI) calculations which demonstrated that prelimi-

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nary layouts of the piping systems in Table 3.2 (LBB piping systems) will satisfy the acceptance criteria in the PEDs:

- ABB-CE Calculation No. ALWR-ME-C-006 "System 80+ Surge Line Routing, Stress Analysis and LBB Evaluation," Revision 01, April 3, 1993.
- DESI Calculation No. 4248-04-1627.00-0001, "Direct Vessel Injection Line For ABB-CE System 80+ ALWR," January 30, 1993.
- DESI Calculation No. 4248-04-1627.00-0002, "Piping Analysis for Shutdown Cooling System for ALWR Plant," March 11, 1993.
- DESI Calculation No. 4248-04-1627.00-0004, "Main Steamline for ABB-CE System 80+ ALWR," January 10, 1993.

Subsequently, CESSAR-DC Section 3.9.2.5, Amendment T, provided clarification that in the stability analyses, the normal operating and seismic loads are summed absolutely and in the leak rate analyses, individual normal load components are summed algebraically. This is acceptable.

On the basis of these reviews, the staff finds that ABB-CE demonstrated that compliance with the PED acceptance criteria can be reasonably achieved by applying customary layout procedures to the LBB piping systems.

During review of the DESI calculations, 4248-04-1627.00-0001, -0002, and -0004, the staff found that in piping systems containing both ASME Code Section III, Class 1 and Class 2 portions of piping, locations of highest stresses were based on ASME Code Class 1 piping criteria. This was acceptable.

Relative to the evaluation criteria in the preceding, the staff found the following:

- (1) In conformance with evaluation criterion (1), normal operating procedure (NOP) loads are defined in CESSAR-DC Appendix 3.9A, Section 1.9.4.4, to include pressure, deadweight, and normal steady-state thermal conditions. NOP loads are used to calculate a crack length that will result in 10 times the detectable leakage rate of 1.0 gpm. This section also specified that NOP loads, critical thermal transients (including loads due to thermal stratification), SSE loads, and NOP dynamic transient loads (such as from rapid valve closure) are considered in the flaw stability analyses. In particular, for the SL, loads due to thermal stratification with a temperature difference of 200 °C (360 °F) were conservatively considered in combination with the limiting plant forced shutdown condition (i.e., reactor trip followed by plant cooldown condition). The largest of the design loads is referred to as the "maximum design" load.

In addition, since the flaw stability analyses are uniformly based on the most limiting material properties (i.e. base-metal tensile properties and the lower of base metal/weld metal fracture toughness properties) and the PEDs are assessed at the locations of highest stresses in the piping systems, the LBB evaluations are bounding for the limiting location in the piping system.

- (2) For ABB-CE facilities there is no history of cracking failures in the MCL and interconnected systems and MSL piping. These piping systems have an operating history which demonstrates their inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer or fatigue (low and high cycle). CESSAR-DC Section 3.6.3.1 provides evaluations to verify that these systems satisfy the applicability requirements of NUREG-1061, Volume 3 for LBB applications. The ABB-CE operating history totals more than 212 reactor-years of operation by 15 plants, including 7 plants each having more than 15 years of operation and 11 plants each with more than 10 years of operation. The ABB-CE operating history includes a total of 21 reactor-years of operation by three System 80 reactor units in the Palo Verde Nuclear Generating Station (PVNGS).

To date, the staff has approved LBB applications in ABB-CE plants for only the MCL hot- and cold-leg piping in three units of the PVNGS, the only System 80 plant. The MCL piping in these System 80 units are similar in size, materials, and methods of construction to the System 80+ designs. Additionally, the staff has approved LBB applications in stainless steel piping systems similar to the System 80+ SL, DVI, and SC piping systems. However, the application of LBB to the MSL piping has not previously been approved by the staff in PWR plants. Accordingly, particular attention was paid to the MSL piping during the staff's evaluation of the applicability of LBB to piping systems in the System 80+ design. The results of this evaluation are discussed next.

In response to issues raised by the staff, ABB-CE revised CESSAR-DC Section 3.6.3.1 to provide additional information regarding the susceptibility of the MSL to: (1) erosion, erosion/corrosion and

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erosion/cavitation, (2) water hammer, (3) indirect causes, and (4) fatigue. As discussed below, the staff finds this information acceptable.

ABB-CE stated that SA-516 Grade 70 carbon steel used for MSL piping was not susceptible to erosion-corrosion under the MSL dry steam operating condition. ABB-CE also stated that for these service conditions, a protective oxide film is formed on the carbon steel piping surface which inhibits corrosion; consequently, the piping is also resistant to erosion/corrosion effects. The staff agrees with these assessments, that is, for piping systems in these service conditions, carbon steel is the preferred material. Carbon steel MSLs have a successful and extensive operating history (thousands of years) in nuclear (PWRs and BWRs) and fossil power plants. Furthermore, considering the lack of experience with the use of alternate materials, the use of such materials could introduce currently unidentified issues.

As a part of the assessment of the potential for adverse effects due to water hammer, ABB-CE provided an analysis of eight documented water hammer events in NUREG/CR-2059 and EPRI NP-6766 (U.S. NRC, "Compilation of Data Concerning Known and Suspected Water Hammer Events in Nuclear Power Plants, CY 1969 - May 1981," NUREG/CR-2059, May 1982 and Electric Power Research Institute, "Water Hammer Prevention, Mitigation, and Accumulation, Volume 1: Plant Water Hammer Experience," Final Report, EPRI NP-6766, July 1992). Of these eight events, six were severe and occurred in piping adjacent to the main steam isolation valves due to valve operation (3 events) and steam-propelled water slugs (3 events). The remaining two were not considered severe and were of unknown causes: one occurred in piping downstream of the steam bypass valve and the other in the main steam relief valve piping. ABB-CE noted that none of these events caused damage to these piping lines. These lines are not part of the MSL for which ABB-CE requested LBB approval. ABB-CE further noted that water hammer due to steam-propelled water slugs were to be prevented by proper procedures for operation of the MSL. It is well recognized that water hammer events in plants usually occurred during the initial stages of operation and reoccurrences were prevented by proper procedures for operation developed in response to these early events. Application of procedures developed for the System 80 plants to the System 80+ design provides adequate assurance that water hammer events in the System 80+ design will be prevented or minimized (see discussion of GSI A-47 in Chapter 20 of this report). The severe water hammer events were isolation-valve-related and produced effects which can be predicted by analysis and have

been considered in the design of the MSL. Moreover, the potential for unanalyzed severe loading events is extremely unlikely considering that no other initiating mechanism has been identified. Finally, the effects of severe, isolation-valve-related events in the MSL will have a lesser impact on the portion of the MSLs inside the containment where LBB is requested for the System 80+ design.

ABB-CE also stated that as an example of prevention of failures due to indirect causes, the overhead polar crane, although not seismic Category I, is designed such that an SSE will not cause failure in a manner to adversely affect a safety system.

Furthermore, in evaluating potential detrimental effects due to fatigue, ABB-CE clarified that because the MSL piping is not subjected to severe thermal or other transient loading, the allowable stress utilized in the analysis of ASME Code, Section III, Class 2 piping (such as the MSL) would include consideration of the number of equivalent full temperature cycles for determining the stress range reduction factors for cyclic conditions. In addition, the code requirements on piping material toughness should be adequate to accommodate dynamic strain aging effects. The staff agrees that this method of fatigue analysis and code requirements for piping material toughness is appropriate for ASME Code, Section III, Class 2 piping systems, including the MSL piping. ABB-CE also noted that fatigue damage in the MSL will be due primarily to thermal cycling during heatup and cooldown and, as indicated in CESSAR-DC Section 3.6.3.1.8, the cyclic thermal stresses are conservatively considered in the ASME Code, Section III, Class 2 design of the MSL. Specifically, for the range of expected equivalent full temperature cycles in the MSL over the life of the plant, the ASME Code Class 2 allowable thermal stress range during these cycles is less than the ASME Code Class 1 allowable range. Subsequently, ABB-CE incorporated this information into Amendment U to CESSAR-DC Section 3.6.3.1.8, which is acceptable.

On the basis of the preceding discussion, the staff finds the application of LBB to the MSL in the System 80+ design is acceptable.

- (3) The material tensile and fracture toughness properties used in the analyses are described in CESSAR-DC Appendix 3.9A, Section 1.9.6.3.

The tensile properties were based on data from the Piping Fracture Mechanics Data Base (PIFRAC), NUREG/CR-4894 (U.S. Nuclear Regulatory Commission, "A Users' Guide to the NRC's Piping Fracture

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Mechanics Data Base (PIFRAC)," NUREG/CR-4894, May 1987). Data used utilized in the piping flaw finite element analyses are shown in CESSAR-DC Figures 3.9A-18 and 3.9A-20 for SA-516 Grade 70 carbon steel and SA-312 Type 316 stainless steel materials, respectively. Corresponding Ramberg-Osgood representations of these data are shown in CESSAR-DC Figures 3.9A-19 and 3.9A-21, respectively. Use of these PIFRAC tensile property data is acceptable.

The material resistance curve for the SA-516 Grade 70 carbon steel piping systems was based on PIFRAC data for a SA-516 Grade 70 shielded metal arc weld as shown in CESSAR-DC Figure 3.9A-22. However, the material resistance curve for the SA-312 Type 316 stainless steel piping systems was based on PIFRAC and the following industry data for gas tungsten arc welds (GTAWs) as shown in CESSAR-DC Figure 3.9A-23:

- Electric Power Research Institute, "Evaluation of the Toughness of Austenitic Stainless Steel Pipe Weldments," June 1986.
- U.S. Nuclear Regulatory Commission, "Analysis of Experiments on Stainless Steel Flux Welds," NUREG/CR-4878 (BMI-2151), April 1987.
- U.S. Nuclear Regulatory Commission, "Short Cracks in Piping and Piping Welds," NUREG/CR-4599 (BMI-2173), April 1992.

ABB-CE specifies in CESSAR-DC Section 3.6.3.4 that SA-312 Type 316 stainless steel piping to be used in piping systems to which LBB evaluations are to be applied will be seamless pipes. In addition, CESSAR-DC Appendix 3.9A, Section 1.9.6.3.2, states that shop and field welds in such piping will be GTAWs. Accordingly, use of these PIFRAC and industry material resistance property data is acceptable.

- (4) Consistent with RG 1.45, the System 80+ MCL pressure boundary has leak detection systems inside the containment boundary such that a leakage of 3.8 L/min (1 gpm) can be detected within 1 hour. The calculated leak rate through the postulated flaw is large relative to the staff's required sensitivity of the plant's leak detection systems. As specified in CESSAR-DC Section 3.6.3.3. and Appendix 3.9A, Sections 1.9.4.2, 1.9.5.2, and 1.9.6.2, the margin is a factor of 10 on leakage and is consistent with the guidelines of NUREG-1061, Volume 3.

The RCPB leakage detection systems for System 80+ are described in CESSAR-DC Section 5.2.5. The staff concludes in Section 5.2.5 of this report that the RCPB detection systems with the detection capability of 3.8 L/min (1 gpm) satisfy RG 1.45 and SRP Section 5.2.5, and therefore, are acceptable.

The MSL leakage detection for System 80+ is addressed in CESSAR-DC Section 3.6.3.3.1. The unidentified plant leakage is continuously monitored using containment sump, reactor cavity sump, and containment cooler condensate tank instruments. An alarm of a leak rate greater than 3.8 L/min (1 gpm) will require the operator to perform a reactor coolant system (RCS) inventory balance, which quantifies the unidentified RCS leakage. Subtracting the RCS unidentified leakage and any known high-energy line leaks inside the containment from the plant leak rate provides an estimate of MSL leakage. An MSL leak rate in excess of 38 L/min (10 gpm) would require plant shutdown according to the Technical Specifications (TS).

The staff has reviewed the proposed MSL leakage detection and compared it to the RCPB leakage detection. The staff finds that the method of MSL leakage detection along with the commitment in the TS is acceptable. The leakage rate of 38 L/min (10 gpm) is based on ABB-CE's calculation of the maximum leakage rate from a crack opening under NOP loadings. ABB-CE calculated this maximum leakage rate to be approximately 300-580 L/min (75-145 gpm). There is sufficient margin by comparing the calculated 300-580 L/min (75-145 gpm) with the TS limit of 40 L/min (10 gpm). Therefore, the staff concludes that the MSL leakage detection for application of LBB to the MSLs in the System 80+ design is acceptable.

- (5) The acceptance criteria in the PEDs were developed in part by applying a margin of $\sqrt{2}$ on loads utilized in the leakage-size flaw stability analyses. This margin of $\sqrt{2}$ on loads is consistent with NUREG-1061, Volume 3 criteria for the method of combining the load components and hence acceptable.
- (6) Similar to Item 5, the acceptance criteria in the PEDs were also developed on the basis of stability analyses of flaws 2 times the leakage-size flaws for the loads considered. This margin of 2 on leakage-size flaw is consistent with NUREG-1061, Volume 3 criteria and hence acceptable.

Relative to the development of the PEDs in Items 5 and 6 (above), the staff would note that the boundaries of

the regions of allowable loading in these diagrams were developed on the assumption that the boundaries were piecewise linear. Additional calculations performed by ABB-CE verified that this assumption was essentially valid and also conservative. Moreover, torsional moments were not considered in the flaw stability evaluations since their effects were small relative to those due to bending moments. Furthermore, two evaluation diagrams are needed for the SL since thermal stratification effects are recognized as significant for the SL. Similar effects in the DVI line were small relative to the other loads and were neglected.

On the basis of its review, the staff finds that the System 80+ MCL, SL, DVI line, SC line, and MSL piping inside the containment comply with GDC 4 according to the criteria in NUREG-1061, Volume 3.

3.6.3.5.4 Conclusions

On the basis of its review, the staff finds that the ABB-CE LBB approach complies with GDC 4 and the Commission approved staff positions on LBB applications as stated in SECY-93-087. Appropriate bounding limits were established using preliminary analysis results, and conformance with such bounding limits will be verified during the COL phase by performing the appropriate ITAAC. Thus, the likelihood of large pipe breaks occurring in the CE System 80+ MCL, SL, DVI line, SC line, and MSL piping inside the containment is sufficiently low so that dynamic effects associated with postulated pipe breaks need not be included in the design basis.

3.7 Seismic Design

Using SRP Section 3.7 as the guideline, the staff reviewed CESSAR-DC Section 3.7 concerning the seismic design adequacy of the ABB-CE System 80+ standard plant. In addition, the staff conducted design audits at Duke Engineering and Services (DE&S) office in Charlotte, North Carolina, ABB-Impell office in San Ramon, California (consultant to ABB-CE), ABB-CE's office in Windsor, Connecticut, and the NRC office in Rockville, Maryland. The audit at the DE&S office covered the soil-structure interaction (SSI) methodology for the NI, including the NA structures, and the design criteria and analysis methodology for the non-nuclear island (NNI) structures. The audit at the ABB-Impell office covered the seismic analysis and modeling issues for the NI structures and included the audit of design calculations for seismic SSI analysis and the development of seismic models. The audit at the ABB-CE office technically resolved or closed a number of seismic modeling and analysis issues noted in the DSER and subsequent design audits. The audit at the NRC office

resolved the issue of System 80+ steel containment vessel (SCV) sliding and overturning during a seismic event, in addition to a number of seismic modeling and analysis issues.

At the April 27 through 30, 1992, meeting, the staff asked ABB-CE to make extensive modifications to CESSAR-DC Section 3.7 because responses to RAI Q220.3 through Q220.40 were extensive and needed to be included in the CESSAR-DC. In the DSER, the staff identified Confirmatory Item 3.7-1 for ABB-CE to revise the CESSAR-DC to include responses to these RAIs and supplemental information related to seismic design.

A significant amount of new information introduced at this was subsequently included in the CESSAR-DC, including:

- In Amendment N to the CESSAR-DC, ABB-CE considered three control motions, CMS1, CMS2, and CMS3 at ground surface and rock outcrop levels as discussed in CESSAR-DC Section 2.5.
- In Amendment N to the CESSAR-DC, ABB-CE included a list of the seismic Category I SSCs in Table 3.2-1. The NI structures are modeled as lumped mass sticks and their typical sketches are shown in CESSAR-DC Figures 3.7-13 through 3.7-17. These models are based on the new plant layout which consists of interconnected containment shield building and NA structures on a common rectangular basemat.

On this basis, DSER Confirmatory Item 3.7-1 is resolved.

3.7.1 Seismic Design Parameters

The seismic input for the design of the System 80+ plant consists of three control motion spectral shapes all anchored at 0.3g horizontal peak ground acceleration (PGA) (Brinkman, C.B. (ABB-CE), letter to NRC, LD-92-051, April 15, 1992; I.M. Idriss for ABB Impell Corporation, "Earthquake Ground Motions - Selection of Control Motion and Development of Generic Soil Sites," Draft No. 2, September 4, 1990; U.S. Nuclear Regulatory Commission, Presentation material from the April 27-30, 1992, NRC/ABB meeting at the NRC office, Bethesda, Maryland: (a) Selection of Control Motion for the System 80+ Standard Design; and CESSAR-DC Section 2.5.2). These three motions are

- (1) RG 1.60 design ground spectra, identified as CMS1, specified at the ground surface in the free field for both deep soil and rock sites. This motion is anchored to a PGA of 0.3g for the two horizontal directions and the vertical direction.

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- (2) The control motion, identified as CMS2, presented in CESSAR-DC Section 2.5.2: these spectra are applied at the rock outcrop. This motion is anchored to a PGA of 0.3g for the two horizontal directions and 0.2g for the vertical direction.
- (3) Another control motion, identified as CMS3, based on NUREG/CR-0098 (Newmark, N.M., and Hall, W.J., "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," NUREG/CR-0098, prepared for the U.S. NRC, May 1978) median spectra: these spectra are specified at the rock outcrop and are modified by shifting the spectral shape for high-frequency cutoff from 8 Hz to 15 Hz. This motion is also anchored to a PGA of 0.3g for the two horizontal directions and 0.2g for the vertical direction.

The response spectra for the input time histories and the associated ground surface and foundation-level response spectra, as well as the time histories matching the target response spectra for CMS1, CMS2, and CMS3 are presented in CESSAR-DC Sections 2.5.2 and 3.7.1. The average power spectral densities (PSDs) of the synthetic time histories are shown in CESSAR-DC Appendix 2B. The control motion spectra (CMS1, CMS2, and CMS3) are also evaluated in Section 2.5.2 of this report.

For the CMS2 case, the response spectra corresponding to the synthetic time histories match the CMS2 rock outcrop response spectra for damping ratios of 5 percent or less (ABB-CE's response to Q220.9, Brinkman, C.B., letter to NRC, LD-92-030, February 25, 1992). By DSER Open Item 3.7.1-1, the staff expressed the concern that the response spectra corresponding to the synthetic time histories do not satisfy the acceptance criteria of SRP Section 3.7.1 for 7-percent damping, the damping ratio associated with reinforced concrete and bolted steel structures under SSE conditions. Figures 1d, 2d, and 3d of RAI Q220.9 (LD-92-030) stated that there are more than five points below the target spectrum for each of the three components of the time histories, H1, H2, and V. In response to this open item, ABB-CE stated in a letter dated December 23, 1992 (LD-92-123), that the CMS2 rock outcrop spectra are below the target spectra in the 3- to 20-Hz range. However, for the seismic SSI analysis, the conservative approach of convolving rock outcrop motion to the surface was used, resulting in high amplitudes of the surface and foundation spectra, instead of the traditional approach of deconvolving the control motion from the ground surface to the rock level. In addition, ABB-CE developed target PSD for CMS2 and CMS3 motions, and demonstrated that PSD for CMS2 and CMS3 are greater than the target PSD throughout the range of frequencies of significance. The staff agrees that

ABB-CE's approach is conservative and, therefore, concludes that the synthetic time histories are acceptable. On this basis, DSER Open Item 3.7.1-1 is resolved.

By DSER Open Item 3.7.1-2, the staff noted that the time histories associated with CMS1 and CMS3 could not be reviewed because they had not been submitted. CESSAR-DC Amendment P, Figures 3.7-1 to 3.7-3 provided the response spectra corresponding to the synthetic time histories for the CMS1 motion along with the target spectra for 2-percent, 5-percent, and 7-percent damping. CESSAR-DC Amendment P, Figures 3.7-10 to 3.7-12, provided the response spectra corresponding to the synthetic time histories for the CMS3 motion along with the target spectra for 1-percent, 2-percent, 5-percent, and 7-percent damping. CESSAR-DC Amendment N, Appendix 2B, provided the PSDs for the CMS1 and CMS3 motions. On the basis of its review of these spectra, the staff concludes that the response spectra for the CMS1 and CMS3 synthetic time histories envelope the target spectra for the corresponding damping ratios and satisfy the guidelines of SRP Section 3.7.1. This resolves Open Item 3.7.1-2.

On the basis of the preceding discussion, the staff concludes that the SSE input ground motions (the design response spectra and the design time histories) documented in CESSAR-DC Sections 2.5.2 and 3.7.1 meet the SRP and RG guidelines and, therefore, are acceptable.

Damping ratios used in the analysis of the CE System 80+ seismic Category I structures comply with the SSE damping ratios specified in RG 1.61. For soils, damping values (soil material damping and energy dissipation as a result of wave propagation) are determined on the basis of soil shear strains induced in the free field. The approach for soil damping complies with the SRP Section 3.7.2 guidelines and is, therefore, acceptable.

By DSER Open Item 3.7.1-3, the staff noted that ABB-CE presented a cursory description of procedures to be used for selecting the percentage of critical damping. It was not clear from CESSAR-DC Table 3.7-1 whether ASME Code Case N-411-1 damping is applied to piping analysis with limitations as described in RG 1.84. CESSAR-DC Section 3.7.1.3, Amendment N, described procedures to be used for selecting the percentage of critical damping. In a reference to CESSAR-DC Table 3.7-1, ABB-CE stated that when response spectra method of analysis is used for piping, damping values may be based on Code Case N-411-1 as limited by RG 1.84. On the basis of ABB-CE's compliance with RG 1.84, the staff concludes that the damping values used in the seismic analysis of seismic Category I SSCs comply with the guidelines of SRP Section 3.7.1. This is acceptable and resolves DSER Open Item 3.7.1-3.

The System 80+ seismic Category I structures have concrete foundations supported on soil, rock, or compacted backfill. CESSAR-DC Section 3.8.5 specifies that the NI foundation mat is approximately 3 m (10 ft) thick, and the top of the NI basemat is located $12.42 \text{ m} \pm 0.3 \text{ m}$ ($40.75 \text{ ft} \pm 1 \text{ ft}$) below the finished grade elevation.

As discussed above, ABB-CE meets the requirements of designing the standard plant to withstand an earthquake that is more severe than the SSE for all currently licensed plants with the exception of the two coastal California sites.

GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes and the design of these SSCs shall reflect appropriate consideration of the most severe of the natural phenomena historically reported. ABB-CE meets the relevant requirements of GDC 2 by appropriate consideration of the most severe earthquake to which the System 80+ Category I SSCs will be subjected, but with the elimination of OBE from design consideration as discussed in Section 3.1.1 of this report. ABB-CE meets these requirements by using the following methods and procedures.

ABB-CE used three sets of ground response spectra to design System 80+ SSCs. Site acceptance criteria are developed to ensure that the design has considered the most severe earthquake for a site selected by the COL applicant. The specific percentage of critical damping values used in the seismic analyses are in conformance with RG 1.61 and ASME Code Case N-411-1, as endorsed by RG 1.84. The synthetic time histories used for the design of System 80+ SSCs were developed in accordance with the guidelines of SRP Section 3.7.1 and were adjusted in amplitude and frequency content to obtain response spectra that envelope the design ground response spectra and also exhibit sufficient energy in the frequency range of interest. Conformance with the recommendations of SRP Section 2.5.2 and RG 1.61 ensures that the seismic inputs to seismic Category I SSCs are adequately defined so as to form a conservative basis for the design of such SSCs to withstand seismic loading.

Certain site-specific SSCs for the System 80+ that are not part of the certified design may be designed using site-specific spectra. To develop these spectra, the horizontal and vertical free-field ground surface site-specific ground motion spectra for the controlling earthquake(s) for a site should be obtained using the procedures of SRP Section 2.5.2. The maximum spectral amplitude of these spectra in the frequency range 5 to 10 Hz should be obtained. Both the horizontal and vertical design response spectra for the System 80+ certified design should be

scaled throughout their entire frequency range in such a manner that the minimum spectral amplitudes of the certified design spectra are equal to the maximum amplitudes of the horizontal and vertical site-specific ground motion spectra, respectively, in the 5- to 10-Hz frequency range. The resulting design response spectra should be used as the minimum for the design of site-specific SSCs for the System 80+ as discussed in Section 2.5.2.5.4 of the CESSAR-DC.

3.7.2 Seismic System Analysis

3.7.2.1 Nuclear Island Structures

The scope of the staff's review of the seismic analysis of the NI structures includes the seismic analysis methods and acceptance criteria used for the System 80+ NI structures. On the basis of CESSAR-DC Table 3.2-1, and Section 3.7.2, the System 80+ RB and NA seismic Category I structures can be grouped as follows.

- reactor building
 - containment shield building
 - SCV
 - internal structure (IS)
 - subsphere
- NA structures
 - control area
 - EFW tank/MSVH areas
 - emergency diesel generator areas
 - CVCS/Maintenance Area
 - spent fuel pool area

The containment shield building, SCV, IS, subsphere, and NA structures are on a common basemat and, thus, form the NI. Before the issuance of the DSER, the staff stated in RAI Q220.13 that ABB-CE did not cover all seismic Category I structures in this section of the CESSAR-DC. The staff asked ABB-CE to submit a list of all seismic Category I SSCs and to identify the method of seismic analysis for each. In response (Brinkman, C.B. (ABB-CE), letter to NRC, LD-92-016, February 12, 1992), ABB-CE referred to CESSAR-DC Table 3.2-1 for the list of seismic Category I SSCs and to CESSAR-DC Section 3.7.3.5 for the analysis methods. ABB-CE's response is sufficient.

For seismic analysis, the NI structures are modeled as lumped mass stick models shown in CESSAR-DC Figures 3.7-13 through 3.7-20. For the NI structures, the seismic SSI analysis is performed using the substructure method formulated in the frequency domain using the

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complex response method and the finite element technique. These modeling and analysis techniques are in accordance with the guidelines of SRP Section 3.7.2 and are, therefore, acceptable.

Appendix 3.7B to the CESSAR-DC includes seismic analysis results for the design of structures and equipment for all soil cases and control motions CMS1, CMS2, and CMS3. Representative plots of envelope SSE response spectra are shown in CESSAR-DC Figures 3.7B-7 to 3.7B-42 and seismic profiles for floor accelerations, cumulative stick shears, axial forces, torques and moments are shown in Figures 3.7B-58 to 3.7B-102. Soil amplification at frequencies in the range of dominant structural frequencies are presented and discussed in CESSAR-DC Appendix 3.7B.

The envelope of the floor shears, axial forces and moments from all soil cases are used to design the NI structures, except for the SCV. In order to compute the distribution of applied loads and to develop design loads for individual shear walls and floor slabs, ABB-CE developed a detailed three-dimensional (3-D) finite element model (FEM) of the NI structures. This model uses shell elements to model the shear walls and floor diaphragms, solid elements to model thick walls and the basemat, and beam elements to model the columns. The FEM is used for load distribution for all structures except for the SCV. ABB-CE has performed benchmark tests to verify proper computation of design forces from the FE analysis results. CESSAR-DC Appendix 3.8A states that the FEMs are used for global in-plane distribution of forces and the out-of-plane design calculations are performed based on local models, which are more detailed, or hand calculations with proper assumptions on end conditions and load distribution. Because the SCV is modeled in the seismic SSI analyses with a simpler model, a special procedure is followed to compute the stresses in the SCV due to the seismic loads. A dynamic analysis of the detailed FEM subjected to the ± 15 percent peak broadened translational and rotational spectra at EL. 28 m (91.75 ft) of the IS (SCV support location) is performed in order to determine the stresses in the SCV.

The CESSAR-DC designates the major NI structures such as the shield building, SCV, ISs, and the NA structures as "seismic systems." A smaller set of safety-related SSCs housed within the major buildings and structures are designated as "seismic subsystems." The decoupling criteria for subsystems is in accordance with the guidelines of SRP Section 3.7.2 and, therefore, is acceptable.

The safety-related structures are modeled as multi-degree of freedom systems using an adequate number of masses. The major structural elements such as slabs, foundation mat, roof slabs, shear walls, and braced vertical frames are

included in the model. All subsystems such as equipment and piping are considered in accordance with the decoupling criteria of SRP Section 3.7.2. For all seismic analyses, dead load plus 25 percent of live load is assumed to contribute to the inertial masses. In addition, mass associated with all heavy equipment is also included in the computation of floor masses.

The modeling approach for all NI structures, except for the SCV, consists of developing equivalent 3-D lumped parameter stick models based on 3-D FEM. Because of its slenderness, the SCV has significant membrane type behavior and is thus modeled using shell elements.

Before the issuance of the DSER, many of the RAI questions (Q220.12, 220.14, 220.16 through 220.20, 220.22, 220.23, 220.27, 220.31, and 220.33, Wambach, T.V., NRC letter to E.H. Kennedy, ABB-CE, September 26, 1991) were associated with various structural models presented in CESSAR-DC Section 3.7.2. In the DSER, the staff noted that it was not completely clear how the model was actually fine-tuned. The staff further noted that a change in nodal masses will alter frequencies and participation factors of all modes at the same time. The staff noted in the DSER that the detailed process of developing dynamic models of NI structures, including the fine-tuning, be documented systematically in an auditable form. The staff also requested that ABB-CE clearly describe this process in the CESSAR-DC. This was DSER Open Item 3.7.2-2. Furthermore, the staff noted that ABB-CE had committed to revising CESSAR-DC Section 3.7 to incorporate the appropriate materials presented in the RAI responses and at the meetings. Subsequently, as described below, ABB-CE submitted extensive explanations in both its responses (LD-92-016) and at the April 27 through 30, 1992, audit (U.S. Nuclear Regulatory Commission, Presentation material from the April 27 through 30, 1992, NRC/ABB meeting at the NRC office, Bethesda, Maryland), March 17 and 18, 1993, audit (U.S. Nuclear Regulatory Commission, Audit summary for the March 17 and 18, 1993, NRC audit, Duke Engineering Services Company office, Charlotte, North Carolina), and June 8 through 10, 1993, audit (U.S. Nuclear Regulatory Commission, Audit summary for the June 8 through 10, 1993, NRC audit, ABB-Impell office, San Ramon, California) as well as in the CESSAR-DC. ABB-CE indicated that the dynamic analysis models of the NI structures were developed by constructing and analyzing detailed FEMs to obtain the dynamic characteristics of these structures. To reduce the number of dynamic degrees of freedom (DDOF) for inclusion in the SSI analysis of the NI, stick models were subsequently developed by separately analyzing the shear walls of two adjacent floors of the detailed FEM to yield equivalent beam properties for each stick. The stick models were then fine-tuned by adjusting the

cross-sectional properties, that is, stiffness, so that the dynamic characteristics of the stick models would match those of the detailed FEMs. In the June 8 through 10, 1993, audit with the staff, ABB-CE stated that the physical mass of the system represented by modal masses was not altered. The staff reviewed the mass calculations and found them acceptable. ABB-CE also stated that the masses used in the development of the stick models were based on a realistic best estimate of masses expected at the time of a seismic event. ABB-CE further stated that this would include the building deadweight due to concrete and steel; major permanent equipment such as tanks, pumps, Hxs, and nuclear steam supply system (NSSS); and an allowance of 2.4 to 3.6 kPa (50 to 75 psf) for commodities such as switchgear, motor control centers, piping and miscellaneous floor loads. The allowance of 2.4 to 3.6 kPa (50 to 75 psf) for live load is approximately equivalent to 25 percent of the floor live load of 9.6 to 14.4 kPa (200 to 300 psf), and is in accordance with the industry practice. ABB-CE also stated that the masses used in the equivalent static FEM analysis and the design are the full live load of 9.6 to 14.4 kPa (200 to 300 psf) in addition to those included in the dynamic model. In CESSAR-DC Section 3.7.2.3.4 (Amendment S) ABB-CE stated that for all seismic analyses, 25 percent of the live load in addition to the dead load is assumed to contribute to the inertial forces. In CESSAR-DC Section 3.7.2.3.4 (Amendment S), ABB-CE further stated that all heavy equipment is also included in the computation of floor masses.

The dynamic model of the steel containment was developed in a similar process except that a coarse-mesh FEM resulted in lieu of a stick model. On the basis of accepted industry practice, the staff finds that ABB-CE's process for developing a dynamic analysis model of the NI is acceptable.

During the June 8 through 10, 1993, audit, the staff audited the detailed process of developing the dynamic models of the NI structures, including the fine tuning to determine the cross-sectional properties of the stick models. The staff found that the tuning of the IS, which includes the subsphere region, was done by constructing a 3-D FEM of the structure and developing stick models which capture the mode shapes and natural frequencies of the major modes of vibration, as described in the CESSAR-DC. The fine tuning was done by adjusting the stick model cross-sectional properties to match the frequencies, mode shapes, and participation factors with the 3-D FEM. The concrete shield building stick model was tuned to capture the dominant frequencies and mode shapes determined from an axisymmetric model, as described in the CESSAR-DC. The cross-sectional areas were reduced in the dome region to account for the vertical displacement

caused by the bending of the dome. Similarly, the shear areas and the moments of inertia were adjusted until a good frequency and participation factor match was obtained for the shield building.

The procedure for the adjustment of the IS, concrete shield building, and NA structures stick models in order to dynamically tune the stick models to 3-D FEMs is incorporated in CESSAR-DC Section 3.7.2.3.4.1.

In the June 8 through 10, 1993, audit meeting with the staff, ABB-CE further stated that the masses of the stick models and the 3-D FEMs differ by less than 1 percent. In addition, the stiffness equivalency was checked by applying unit loads on a sample of major floors, both for the stick and the FEMs, and calculating the displacements produced by the unit loads. The results showed that the displacements from the stick models are similar to the displacements of the FEMs. On the basis of the matching of the natural frequencies and mode shapes and the correlation between the mass and stiffness, the staff concludes that the stick and the FEMs are dynamically equivalent.

The additional NI structures which comprise the NA were divided into eight areas. FEMs were constructed for each floor and stick model properties were determined from these models. On the basis of its review, the staff determines that the procedure used for modeling of structures in order to properly represent the 3-D characteristics of structures is in accordance with the SRP Section 3.7.2 guidelines and is, hence, acceptable. In addition, the procedures used for designating systems versus subsystems and the decoupling criteria for subsystems as described in the CESSAR-DC are in accordance with the SRP Section 3.7.2 guidelines. The staff, therefore, concludes that the process of developing dynamic models of NI structures is acceptable. This resolves DSER Open Item 3.7.2-2.

In Section 3.7 of the DSER, the staff discussed that the seismic analyses of the NI documented in the CESSAR-DC, Amendment I, were based on old structural layouts. Using the results of these analyses, ABB-CE performed a preliminary buckling assessment of the steel containment. At the April 27 through 30, 1992, meeting with the staff, ABB-CE presented new stick models for the containment shield building and the NA structures which were developed on the basis of new structural arrangements. In the DSER, the staff noted some modeling anomalies. For example, staff questioned how independent sticks can be generated for the case of structures that are connected by large, rigid, horizontal diaphragms supported by vertical shear walls. Large torsional stiffnesses of floor diaphragms are needed to couple these various areas of the structures located on the common foundation basemat.

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Before preparing the DSER, the staff did not conduct a comprehensive review of the modeling to determine whether or not the corresponding sticks were properly coupled to form a finite element system analysis model. Also, the staff noted that the sticks were not connected to the exterior vertical walls of the shield building in the seismic analysis for soil-structure interaction (SASSI) models, although, physically, the diaphragms are rigidly attached to the outside vertical walls. By DSER Open Item 3.7.2-4, the staff noted that ABB-CE should define the criteria to ensure that the stick models developed are equivalent to the 3-D FEMs. The staff also noted that SSI analysis for the NI structures had not been performed on these new stick models.

In a response dated February 2, 1993 (LD-93-011) to DSER Open Item 3.7.2-4, ABB-CE noted that since large and rigid horizontal diaphragms connect the NI structures, the individual stick models are also connected at the appropriate elevations with rigid links. In this manner, the rigidity of the slabs is properly modeled. An additional analysis was performed by ABB-CE (LD-93-011) in order to assess whether connecting the stick models rigidly to the outside vertical walls affects the dynamic response of the models. Soil Case B-3.5 was used as the benchmark case, and it was reanalyzed using a structural model which was connected to the outside vertical walls. Results from this analysis and their comparison to the original results without connection to the outside walls are shown in ABB-CE letter LD-93-011 dated February 2, 1993. These results indicate that the original analysis was conservative because when the stick models are connected to the outside vertical walls, the spectral amplitudes are lower at all frequencies. ABB-CE did additional studies which were presented to the staff during the June 8 through 10, 1993, audit. ABB-CE contended that for the RB and NA complex area, the summation of the individual torsional moments of inertia of all sticks is much less than the torsional stiffness provided by the transfer terms when calculating the torsional stiffness about the torsional center of the floor. Hence, splitting the common walls of two adjacent areas in two stick models has minimal impact on the torsional stiffness of the entire RB and NA complex. In addition, ABB-CE presented analysis results to demonstrate that the torsional response is not significant and that responses in one horizontal direction due to the application of earthquake in the other horizontal direction are low and, therefore, the torsional response does not govern the design forces. In order to demonstrate the dynamic equivalency of the stick models to the 3-D FEMs, ABB-CE compared the mass and stiffness of the two models. ABB-CE reported (June 8 through 10, 1993, audit) that the masses of the stick models and the 3-D FEMs differ by less than 1 percent. Stiffness equivalency was checked by applying unit loads on a sample of major

floor elevations, one floor at a time, and calculating the displacements produced by these unit loads. The same process was used for both the stick and the FEMs and the results showed that the displacements from the stick models were similar to the displacements of the FEMs. On the basis of these evaluations the staff concludes that the procedures used for modeling of systems in order to properly represent the 3-D characteristics of structures are adequate and are in compliance with the guidelines of SRP Section 3.7.2. On this basis, DSER Open Item 3.7.2-4 is resolved.

By DSER Confirmatory Item 3.7.2-1, the staff sought assurance that the relative displacement among supports and the rocking and torsional effects are properly accounted for in the seismic analysis of System 80+ SSCs. Subsequently, ABB-CE revised the text to incorporate the appropriate portions of the responses into CESSAR-DC, Amendment N. This is acceptable and resolves Confirmatory Item 3.7.2-1.

The major components of the RCS are constructed per NCA-1110 of the ASME Code for the Section III, Class I components. These components are the reactor vessel, steam generators, RCPs, the reactor coolant main loop piping, the SL, and the pressurizer. CESSAR-DC Section 3.7.2.1.2 describes the seismic system analysis performed to assess the dynamic interaction between the components of the RCS and between the RCS and the RB and other structures.

The seismic system analysis is performed for a composite model developed by coupling simplified models of (1) the interconnected major RCS components, excluding the SL and (2) the reactor and other building structures and their common basemat. The representation of the reactor vessel assembly in this simplified RCS model includes a simplified model of the internal components of the reactor to assess the dynamic interaction between those components and the vessel assembly. Methods of analysis for this model, including seismic inputs, procedures for damping and the methods for combining modal responses and the responses to the three earthquake components are in accordance with the review criteria of SRP Section 3.7.2 and are, therefore, acceptable.

In RAI Q210.33 (letter from T.V. Wambach (NRC) to E.H. Kennedy (ABB-CE), September 26, 1991), the staff asked ABB-CE to demonstrate that the simplified model for the RCS was in accordance with the guidelines of SRP Section 3.7.2, Revision 2, Paragraph II.1.a.(iii). ABB-CE responded, in LD-92-016, dated February 12, 1992, by explaining that the number of DDOF was twice the number of modes in the model. In the DSER, the staff expressed a concern that ABB-CE did not demonstrate that inaccura-

cies arising from the use of a single mass point for a run of piping, as was the case of the hot leg piping, were adequately addressed. Accordingly, by DSER Open Item 3.7.2-5, the staff requested that ABB-CE demonstrate that issues addressed in SRP Section 3.7.2, Paragraph II.1.a.(iii) on reducing large static models to models with a smaller number of degree-of-freedom for dynamic analysis have been satisfactorily considered in the coupled system analysis model. ABB-CE responded by letter LD-92-123, dated December 23, 1992, that the dynamic modeling of the hot leg piping was based on matching significant frequencies as determined by a multi-mass model for each hot leg. Each hot leg was modeled using 9 mass points with 27 DDOF. ABB-CE further stated that the predominant frequencies are 178.4 Hz and 183.1 Hz in the lateral directions and 483.6 Hz in the axial direction. ABB-CE contended that the use of only 1 mass point with 2 DDOF is sufficient to match the fundamental frequencies in the lateral directions. ABB-CE stated that matching higher frequencies would serve no purpose since inclusion of these modes will not change the response of the hot leg piping to seismic excitation. On the basis of this information, the staff concludes that the modeling of the hot leg in the composite model is adequate and satisfies the SRP Section 3.7.2 recommendations for reducing large models to fewer degrees of freedom (DOF) models for dynamic analysis. This resolves DSER Open Item 3.7.2-5.

The substructuring methodology of the computer program SASSI as described in CESSAR-DC Appendix 3.7B was used for the seismic SSI analysis. In this substructuring method, the soil strata and half-space are analyzed first in the frequency domain. From this analysis, the impedances at the soil-structure interface are computed. Subsequently, the impedances are combined with models of the superstructures; the control motion is applied to the combined system, and the equations of motion are solved for computation of seismic accelerations and displacements. Figures 3.7-21 and 3.7-22 of the CESSAR-DC show schematic diagrams of the SSI analysis procedure. For the CMS1 motion, the control motion is applied directly at the free-field ground surface. For the analysis using the CMS2 and CMS3 motions, the rock outcrop motion is convolved through the soil media to produce the surface and the foundation level motions. The computed surface motion is applied as the control motion in the SASSI SSI model at the free-field ground surface.

A confirmatory SSI analysis of the NI structures was performed by the staff, using the NRC computer program CARES, Version 1.1. The purpose of the confirmatory analysis was to check the accuracy of ABB-CE's results of the SSI analysis of the NI structures obtained by using the large finite element analysis program, SASSI. All of the NI structures (listed earlier in this section) are supported

by one common foundation basemat. In the confirmatory analysis using the CARES Version 1.1, the CMS2 ground motion was applied at the rock outcrop, and convolved up to the bottom of the basemat where the soil springs and dashpots were assumed to act. In the SASSI analysis, the rock outcrop motion was convolved up to the ground surface from where it was deconvolved down during the SSI analysis. The same structural model that was used in the SASSI program was used in the CARES program. However, there was considerable difference in the manner of modeling the effects of the foundation soil medium in the two codes, namely, SASSI and CARES. In SASSI, the effect of the foundation soil stiffness was considered at various nodes along the excavated soil boundary, whereas in the CARES program, the simplified SSI model assumed the soil springs and the dashpots to act at one central point at the bottom of the mat. Both the ground response spectra and the FRS predicted by the CARES program at selected points in the simplified SSI model agreed well with the SASSI results for the soil case B3.5, using the CMS2 ground motion. This confirmatory analysis provides confidence in the accuracy of the SSI analysis results performed by ABB-CE using the SASSI program.

The site ground surface response time histories from the control motions CMS1, CMS2, and CMS3 discussed in Section 2.5 of this report were used as input to the SSI analysis model. ABB-CE used the SASSI computer code to perform SSI analyses (CESSAR-DC Appendix 3.7B). In these analyses, soil properties are derived from the initial free-field computations for each generic soil column model, using either the SHAKE computer code or the initial SASSI segment, SITE. The staff raised a number of questions in the DSER regarding procedures to be implemented to ascertain "bounding" estimates of the responses of the System 80+ SSCs.

The staff's primary concern with ABB-CE's approach was how the analysis results from the 13 generic soil column cases can be demonstrated to bound the dynamic response of multiple-mode response systems of interest. The staff contended that, in general, each soil column develops different amplification at the modal frequencies of these SSI systems, with the system frequencies being different for each soil column case. In addition, a peak system response of interest will be a summation of a few modal responses. The staff further noted that for control motion CMS1 where the motion is defined at the ground surface, the development of "bounding" free-field responses was not clear. Because "bounding" was not demonstrated in the CESSAR-DC, the staff tried (Q220.0, letter from T.V. Wambach (NRC) to E.H. Kennedy (ABB-CE), September 26, 1991) to determine if the COL applicant would perform site-specific SSE analyses to confirm that the System 80+ standard design is indeed sufficient for the

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site. In response (LD-92-016), ABB-CE stated that there was no intent to supply either site-specific SSE analyses or results in the site-specific SAR. As a result, the staff noted in the DSER that ABB-CE may use any justifiable methods to address the issue of "bounding," but that a limited number of sensitivity studies without further justification would not be acceptable. This was DSER Open Item 3.7.2-3.

In response to the above concern, ABB-CE stated (LD-92-123, letter dated December 23, 1992, and the CESSAR-DC) that ABB-CE had selected the three ground motions and the 13 different soil profiles as the seismic basis for the System 80+ with the objective to obtain a conservative envelope for seismic demand. Initially, ten generic soil profiles (including the rock case) were selected to cover a broad band of shear wave velocities and depths to bedrock for the site. These profiles were A-1, B-1, B-2, B-3, B-4, C-1, C-2, C-3, D-1, and the rock case. After applying the CMS2 control motion at the rock outcrop and convolving the motion to the free-field soil surface, the response spectra obtained at the ground surface were enveloped by ABB-CE to determine if any significant "valleys" occurred in the envelope of the original soil cases. ABB-CE then determined that three additional soil cases, B-1.5, B-3.5 and C-1.5, were necessary in order to fill in these valleys and obtain an envelope of surface spectra that covered a broad band of frequencies with no significant deficiency in any frequency range. In addition, the envelope of the spectra at the free-field foundation level showed that no significant valleys occurred at the free-field foundation level. With the addition of the CMS1 and CMS3 motions, the seismic demand further increased, thus adding more conservatism to the response.

In the soil and SSI analyses, the CMS2 and CMS3 motions are applied as rock outcrop motions. To obtain motion at the ground surface or at the foundation level in the free-field, the CMS2 and CMS3 motions are convolved through the soil profiles of each of the 12 soil cases considered. The resulting motions at the free-field surface and foundation level provide higher spectral amplitudes compared to the CMS2 and CMS3 rock outcrop spectra at frequencies close to the soil column frequencies, and in many cases, the PGA at the surface and foundation level is higher than 0.3g for both the horizontal and vertical directions. For deep soil sites, this introduces conservatism, since, per the SRP recommendations, for deep sites the control motions may be applied at the ground surface.

In order to study the effect of the variation of Poisson's ratio on the vertical SSI analyses, ABB-CE performed a study in which the seismic response at key locations such as ground surface was determined at two Poisson's ratios

of 0.3 and 0.47 instead of the value of 0.4 assumed in the CESSAR-DC. The results of this study were presented in a meeting with the staff on July 28, 1993 (U.S. Nuclear Regulatory Commission, Audit summary for the July 28, 1993, NRC audit, NRC office, Rockville, Maryland). ABB-CE found that the vertical response spectra were significantly impacted when the SSI analysis was performed using damping of one-third of that used in the horizontal SSI analysis in compliance with the CESSAR-DC. However, ABB-CE contended that the current thinking of the industry is that the P-wave damping used in vertical SSI analysis should be equal to the S-wave damping used in the horizontal SSI analysis. ABB-CE evaluated the impact of using Poisson's ratios of 0.3 and 0.47, considering the same damping in the P-wave analysis as in the S-wave analysis. The impact of the variation of Poisson's ratio in the vertical direction was not significant when the damping value for the vertical direction was equal to that in the horizontal direction. In addition, ABB-CE contended that regardless of the damping and Poisson's ratio used in the certification application, the site can be considered as qualified as long as the vertical motion at the surface as developed by the COL applicant for site-specific conditions is enveloped by the System 80+ surface spectra included in the CESSAR-DC. On the basis of the results of this study, the staff concludes that the COL applicant must address the impact of the Poisson's ratio and its variability on the vertical seismic response. Subsequently, CESSAR-DC Section 2.5.4, Amendment U, incorporated the above information. This is acceptable. This is included in COL Action Item 2.5-1.

During the audit of June 8 through 10, 1993, the staff raised the issue that the cutoff frequencies used in the SHAKE computations may impact the strain computations. The staff requested that ABB-CE evaluate the impact of the cutoff frequency on the high-frequency computation in the soil column for the soft sites. ABB-CE contended in the July 28, 1993, audit that a cutoff frequency of 40 Hz was used in all SHAKE analyses in order to satisfy the criteria that the cutoff frequency chosen is (1) high enough within the frequency range of interest and (2) consistent with the frequency content of input motions. This is acceptable.

ABB-CE also addressed the staff concern that deep-soil sites of relatively stiff soils may not have been covered in the range of columns investigated by ABB-CE. The ABB-CE study evaluated two additional soil column cases: case D-1 soil column was extended to a depth of 122 m (400 ft), and an additional soil column in which the shear wave velocity varied from 305 m/sec (1000 ft/sec) to 14 m/sec (3000 ft/sec) over a depth of 244 m (800 ft) from the ground surface. The results of the study, presented in the July 28, 1993, meeting with the staff, demonstrated that the surface motions from both of these soil column

cases were within the envelope of the surface spectra used as the design basis for the System 80+.

During the June 8 through 10, 1993, audit, the staff noted that ABB-CE needs to show that the Winkler soil spring and the uniform soil spring derived from a finite element analysis are compatible for a governing SSI case. ABB-CE reanalyzed the Winkler sensitivity model with applied loads from Case B-1 for the CMS1 motion of the SSI analysis. On the basis of the results obtained (July 28, 1993, meeting), the staff concludes that the Winkler representation is adequate for the purposes of static analysis of the superstructure and the basemat.

ABB-CE presented the methodology for the computation of dynamic lateral soil pressure in the audit of June 8 through 10, 1993. ABB-CE stated that two-dimensional (2-D) models of the NI structures, obtained by taking a slice through the middle of the NI structures, are used in the SASSI models and that simple models of the turbine and the radwaste buildings are included to account for the increased pressure due to these buildings. Furthermore, torsional displacements at corners of basemat are evaluated and the lateral pressures are increased by the ratio of horizontal displacement due to torsion to the horizontal displacement due to translation only. The staff concludes that this approach effectively takes into account the effect of adjacent structures as well as the torsional response on the dynamic lateral pressures and is, therefore, acceptable. With regard to the structure-to-soil-to-structure interaction (SSSI), ABB-CE stated (June 8 through 10, 1993, audit) that, due to the weight and size of the RB and NA complex, it expects that SSSI is a consideration for other structures that are located in the vicinity of the NI structures. Two-dimensional FEMs from the lateral pressure evaluation are used to assess the changes in the near-field motion at the surface due to the dynamic influence of the RB and NA complex and the modified free-field surface motions are applied as input to the final analysis of other seismic Category I structures. The staff concludes that this approach is in compliance with the guidelines of SRP Section 3.7.2 for the coupling between the structures and soil and is, therefore, acceptable.

ABB-CE used the time-history method of analysis to generate the FRS in accordance with the guidelines of RG 1.122, "Development of Floor Design Response Spectrum for Seismic Design of Floor-Supported Equipment or Components," Revision 1. The horizontal and vertical models of the NI are decoupled and floor responses in horizontal and vertical directions are obtained by three separate analyses. The effects of floor flexibility on vertical response spectra for floors with frequencies less than 40 Hz are considered as described in CESSAR-DC Section 3.7.2.5. The spectra are generated for appropriate

damping ratios and the peaks are broadened by ± 15 percent and smoothed in accordance with the guidelines of RG 1.122.

ABB-CE has presented the site acceptance criteria in Section 2.5.2.5.3 and Figure 2.5-38 of the CESSAR-DC. For a rock site, site-specific response spectra at 5 percent damping in the horizontal and vertical directions will be developed and compared to the envelope of the CMS1, CMS2, and CMS3 control motions. If the site-specific response spectra are enveloped by the envelope of the CMS1, CMS2, and CMS3 response spectra, the site will be considered acceptable for construction. If the site-specific response spectra exceed the envelope of the CMS1, CMS2, and CMS3 response spectra at any frequency, a site-specific evaluation will be performed. In this evaluation site-specific structural dynamic analysis will be performed and the resulting in-structure response spectra at six critical locations (foundation basemat El. 15.24 m (50 ft), interior structure El. 27.97 m (91.75 ft), control room El. 35.2 m (115.5 ft), top of steel containment vessel El. 76.5 m (251 ft), interior structure El. 44.5 m (146 ft), and shield building El. 80.31 m (263.5 ft)) will be compared to the respective design response spectra, which are the envelope of all generic rock and soil cases. If the in-structure response spectra from the site-specific evaluation are within 10 percent of the in-structure design response spectra, for each of the six locations, the site is considered acceptable. If the in-structure response spectra from the site-specific evaluation exceed the in-structure design response spectra for any of the six locations by more than 10 percent at any frequency, the design might still meet the certified design and licensing commitments due to the substantial design margin between the certified design commitments and the actual bases upon which the plant was designed. To demonstrate that the plant design meets the certified design and licensing commitments, a confirmatory site-specific evaluation will be performed to demonstrate that System 80+ standard certified design meets all the applicable design criteria when subjected to the site-specific response spectra. The results of the confirmatory site-specific evaluation will be reviewed and approved by the staff.

For a soil-site, site-specific response spectra at 5 percent damping in the horizontal and vertical directions at the free-field ground surface will be developed and compared to the envelope of the CMS1 spectra and the surface spectra from CMS2 and CMS3 control motions. If the site-specific ground surface response spectra are enveloped by the envelope of the CMS1 spectra and the surface spectra from CMS2 and CMS3, the site will be considered acceptable for construction. If the site-specific ground response spectra exceed the envelope of the CMS1 spectra and the surface spectra from CMS2 and CMS3 at any

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frequency, a site-specific evaluation will be performed. In this evaluation, in-structure response spectra at six critical locations, defined above for rock site evaluation, obtained from the site-specific evaluation will be compared to the respective design response spectra, which are the envelope of all generic rock and soil cases. If the in-structure response spectra from the site-specific evaluation are within 10 percent of the in structure design response spectra for each of the six locations, the site is considered acceptable. If the in-structure response spectra from the site-specific evaluation exceed the in-structure design response spectra for any of the six locations by more than 10 percent at any frequency, the design might still meet the certified design and licensing commitments due to the substantial design margin between the certified design commitments and the actual bases upon which the plant was designed. To demonstrate that the plant design meets the certified design and licensing commitments, a confirmatory site-specific evaluation will be performed to demonstrate that System 80+ standard certified design meets all the applicable design criteria when subjected to the site-specific response spectra. The results of the confirmatory site-specific evaluation will be reviewed and approved by the staff.

The above actions are referenced in CESSAR-DC Section 3.7, and they are included in COL Action Item 3.7-1. This is acceptable.

The design of the System 80+ subsystems such as piping and equipment is based on one of the four options discussed below. The first option is to use an envelope of broadened response spectra for all rock and soil cases at the appropriate location as input to the analyses of the subsystems. The second option is to group sites into major groups such as hard soil and rock sites, medium soil sites, or soft soil sites. For each group, the envelope of broadened response spectra is developed and used. The third option is to use ± 15 percent peak shifted response spectra from each soil case. The fourth option is to use site-specific response spectra. CESSAR-DC Figure 2.5-38 presents the site acceptance criteria in order to provide a consistent basis for the analysis and design of the System 80+ SSCs. Therefore, when a COL applicant meets the site acceptance criteria, the design will contain all the conservatism embedded in the design of the System 80+, which is based on the 13 generic soil cases. On the basis of the provisions in the analysis, design and site acceptance process described herein, the staff concludes that the 13 generic soil cases provide a conservative basis for the design of the System 80+ SSCs. On this basis, DSER Open Item 3.7.2-3 is resolved.

In the seismic analysis of the NI structures, three statistically independent, orthogonal, components of earthquake

motion are applied to the structural models as separate load cases. For the time-history analysis, the total response is obtained by algebraically adding the responses at each time step caused by each of the three components of earthquake motion. For the response spectrum analysis, the total response is obtained by taking the square root of the sum of squares (SRSS) of the maximum codirectional responses caused by each of the three components of earthquake motion at a particular point of the structure.

For the response spectrum method, the approach to determine the total seismic response of the structure is to combine the response of individual modes of the structure in accordance with the guidelines of RG 1.92 and, therefore, is acceptable.

Before the issuance of the DSER, the staff asked ABB-CE (Q220.24, letter from T.V. Wambach, (NRC) to E.H. Kennedy (ABB-CE), September 26, 1991) to submit the analysis methods and the design criteria that will be applied to the non-safety-related structures to ensure protection of adjacent safety-related structures. ABB-CE committed to revising Section 3.7.2.8 to include three design scenarios for the non-safety-related structures (LD-92-016, February 12, 1992). Under Scenario A, sufficient separation will be maintained between non-safety-related structures and safety-related structures in order to ensure that the failure of a non-safety-related structure due to a seismic event does not impair the integrity of an adjacent safety-related structure. Before the issuance of the DSER, ABB-CE's response was not complete because the analysis methods and the design criteria used to ensure structural integrity of non-safety-related structures during and after the occurrence of the postulated SSE were not provided for Scenario B. In this scenario, when the non-safety-related structure fails, it may fall onto the adjacent safety-related structure. This was DSER Open Item 3.7.2-6. In Scenario C, the safety-related structure is designed to withstand loads due to the collapse of the adjacent non-safety-related structure if sufficient separation of the structures is not achieved. In order to clarify Scenario B, ABB-CE revised CESSAR-DC Section 3.7.2.8 (Amendment Q) to state that the non-Category I structures adjacent to the safety-related structures will be analyzed and designed to prevent their failure under SSE conditions in such a manner that the margin of safety of these structures is equivalent to that of Category I structures. This requirement is in accordance with the guidelines of SRP Section 3.7.2 and is, therefore, acceptable and resolves DSER Open Item 3.7.2-6.

In order to account for the expected variations of structural properties, dampings, and other parameters, the response spectra are generated for appropriate critical damping

ratios and the peaks are broadened by ± 15 percent and smoothed in accordance with the guidelines of RG 1.122.

Before the issuance of the DSER, ABB-CE stated in CESSAR-DC Section 3.7.2.9 that the soil property related spectrum peaks are further broadened, where required, to conservatively account for all potential variations of soil properties within the envelope of site soil conditions. In the DSER, the staff noted that this statement may imply that the floor response spectrum for each soil case will be broadened first to account for uncertainty of soil properties before the spectra of all soil cases are enveloped. Further, the staff noted in the DSER that it was not clear whether or not ABB-CE considered the variation in soil properties by using three sets of values, as recommended in SRP Section 3.7.2. By DSER Open Item 3.7.2-7, the staff requested that ABB-CE revise the text to clearly describe the procedures employed.

According to the response submitted by ABB-CE in the June 21 through 23, 1993, audit (U.S. Nuclear Regulatory Commission, audit summary for the June 21 through 23, 1993, NRC audit, ABB-CE office, Windsor, Connecticut), the effects of parameter variations on FRS are incorporated into the raw spectra in accordance with the four options discussed earlier. The design of piping, components and equipment is based on one of these four options. In addition, ABB-CE performed a parametric study to assess the effects of cracked concrete on the in-structure response spectra. An SSI analysis was performed assuming cracked concrete cross-sections during the SSE event and the reduced stiffness for all concrete structural elements. To account for this stiffness reduction, the Young's modulus of elasticity of concrete was reduced to 70 percent of its original (uncracked) value. Soil Cases A-1, B-3.5 and the fixed base with concrete backfill, all with the CMS2 motions, were selected for the parametric study, since these were controlling soil cases in the SSI analysis as described in CESSAR-DC Appendix 3.7B, Section 2.2, Amendment Q. For concrete structures, such as the IS, it is shown that the effects of concrete cracking caused a peak shifting of less than 15 percent and a small amplitude increase. The increase in amplitude near the predominant spectral peak was below the envelope of all the soil cases. ABB-CE concluded that since several different cases were considered in the SSI analyses, the frequency shift and amplitude change due to concrete cracking is covered by the 15-percent spectra broadening criteria established for the design of piping, components and equipment. For the SCV, concrete cracking made almost no difference in its response.

In order to assess the effects of uncertainties in the material properties of the soil, a parametric study using Case B-3.5 was performed by ABB-CE. Per the SRP

recommendations, low-strain shear modulus variations of $2\frac{1}{2}$ times the best-estimate values must be considered. However, ABB-CE stated that, for Case B-3.5, half of the low strain modulus results in a soil profile that is too soft to support nuclear power plant structures. Therefore, ABB-CE did an upper bound soil analysis with 2 times the low-strain shear modulus using Case B-3.5. For stiffer soil profiles, the low-bound case would coincide with a softer soil case among those analyzed for the System 80+. Therefore, ABB-CE contended that the results of the upper-bound B-3.5 are sufficient to assess the effects of soil variation. As shown in ABB-CE's letter LD-93-006 dated January 20, 1993, a shift of less than 15 percent in the fundamental frequency of response occurred due to the stiffening of the soil. A small increase in amplitude was also observed at the shifted peak. However, this increase was well below the envelope of all the soil cases using the CMS2 motion. For the SCV, there was almost no shift in the fundamental spectral peak, and there was some increase in the amplitude. The resulting amplitude was also well within the envelope of all the soil cases using the CMS2 motion as shown in ABB-CE's letter LD-93-006 dated January 20, 1993.

On the basis of these studies, the staff concludes that variation in soil material properties are adequately considered and the recommendations of SRP Section 3.7.2 for the procedures to consider the effects of the expected variations of structural properties, damping, soil properties, and SSI on the FRS and time histories are satisfied. This resolves DSER Open Item 3.7.2-7.

Constant seismic vertical load factors are not used in the seismic analysis of seismic Category I SSCs. Seismic Category I SSCs are analyzed in the vertical direction using the methods described earlier. The vertical floor flexibilities are accounted for in the response spectra at each individual floor elevation of the building structures.

In the June 8 through 10, 1993, audit, ABB-CE presented the methodology to take into account the effect of vertical floor flexibility on the vertical in-structure FRS. ABB-CE stated that the vertical amplification factors are calculated for the most common slab and support system assuming cracked conditions wherever applicable. The amplification factors also depend on the elevation where the slab is located and are applied as a multiplier to the vertical earthquake zero period acceleration (ZPA) at each elevation.

In the June 8 through 10, 1993, audit, ABB-CE stated that the load combination associated with the buoyant forces due to the maximum groundwater level and the envelope seismic forces produced significant uplift of the NI basemat. However, uplift analysis considering the individ-

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ual SSI analysis cases rather than an envelope of seismic response of all cases showed a minimal uplift. This uplift area was subsequently modeled in the 3-D finite element static model of the NI which was used to predict the maximum stresses used for the design of the basemat. ABB-CE completed the uplift analysis and incorporated the results into CESSAR-DC Appendix 3.7B, Section 3.0, and Appendix 3.8B, Section 6.1.2.

The models used in the seismic analysis of seismic Category I SSCs give 3-D representation of the systems including torsional DOF. In CESSAR-DC Section 3.7.2.11 of early amendments, ABB-CE stated that, in addition to the eccentricity due to the non-symmetrical configuration, an additional eccentricity of 5-percent of the maximum building dimension was included to account for accidental torsion during an earthquake. It was clarified in the April 27 through 30, 1992, meeting that the additional torsion was not incorporated in the dynamic analysis. It was, instead, applied to the static FEMs of structures to calculate element forces and moments. The staff, based on engineering judgment, found this approach acceptable. In the DSER, the staff identified Confirmatory Item 3.7.2-3 that ABB-CE should modify the CESSAR-DC to clearly describe the procedures employed. ABB-CE revised Section 3.7.2.11 of the CESSAR-DC, Amendment N, to state that the additional 5-percent eccentricity are applied to the static finite element structural model to calculate element forces and moments. On the basis of this revision, DSER Confirmatory Item 3.7.2-3 is resolved.

CESSAR-DC Sections 3.7.1.3, 3.7.2.1.2.1, and 3.7.3.6 specify that damping is in accordance with RG 1.61 or CESSAR-DC Table 3.7-1, or both. By DSER Confirmatory Item 3.7.2-2, the staff requested that ABB-CE revise the footnote to CESSAR-DC Table 3.7-1 of the early amendments, which specified that when the response spectrum method of analysis is used, piping damping values may be based on ASME Code Case N-411-1. The staff asked ABB-CE to commit to all of the conditions of RG 1.84 on the use of ASME Code Case N-411-1. Subsequently, ABB-CE revised the footnote in Table 3.7-1 of the CESSAR-DC, Amendment N, to state that the Code Case N-411-1 damping may be used as limited by RG 1.84. On this basis, DSER Confirmatory Item 3.7.2-2 is resolved.

In CESSAR-DC Section 3.7.2.15 of the early amendments, ABB-CE stated that composite damping is to be used for modal superposition methods of analysis and that proportional damping will be used for direct integration time-history methods. In the DSER, the staff required that (1) both composite damping and proportional damping be clearly defined and (2) adequate guidance be provided for estimating the damping ratio for the latter method for the

case where element damping in the various elements of the model are different. This was DSER Open Item 3.7.2-8. Subsequently, in CESSAR-DC Amendment O, ABB-CE provided the procedure for determining the composite modal damping for use in the modal superposition method. According to CESSAR-DC Amendment Q, the time-history method using the proportional damping is only used for subsystems such as piping. In CESSAR-DC Section 3.7.3.15.1, Amendment Q, ABB-CE also presented the methodology for estimating the damping ratios to be used in the proportional damping matrix for the time-history method. For subsystem dynamic time-history analyses using direct integration, damping is input using a proportional damping matrix $[C]$ that is given by:

$$C = \alpha M + \beta K$$

where M is the subsystem mass matrix and K is the subsystem stiffness matrix. The damping ratio ξ at any frequency f_i is given by

$$\xi = \frac{\alpha}{4\pi f_i} + \beta \pi f_i$$

The values of α and β are computed by specifying damping ratio ξ at two frequencies f_1 and f_2 . CESSAR-DC Figure 3.7-33 shows a typical plot of damping value versus frequency. Frequencies f_1 and f_2 are selected so that all subsystem modes with significant participation lie between the frequencies f_1 and f_2 . Furthermore, damping at frequencies f_1 and f_2 are selected from Table 3.7-1 of the CESSAR-DC. In this manner, the damping values for all frequencies between f_1 and f_2 will be less than the selected ratio. Since all significant modes lie between f_1 and f_2 , the damping values for these modes will also be less than the selected damping value. The staff concludes that this procedure provides adequate guidance for estimating the damping values for the composite modal damping and proportional damping methods and is therefore acceptable. This resolves Open Item 3.7.2-8.

In the DSER, the staff identified Confirmatory Item 3.7.2-4 for ABB-CE to revise the CESSAR-DC to either submit detailed methodologies for the analyses of the safety-related dams or commit to present these details in the site-specific SAR. CESSAR-DC Section 3.7.2.13 in conjunction with Section 2.5.5 and Table 1.10-1 state that, if applicable for the site, analyses of safety-related dams will be performed and the appropriate methods will be provided in the site-specific SAR. This is included in COL Action Item 2.5-1. On this basis, DSER Confirmatory Item 3.7.2-4 is resolved.

On the basis of its review, the staff concludes that the plant design is acceptable and meets the requirements of

GDC 2 and Appendix A to 10 CFR Part 100. This conclusion is based on the following:

ABB-CE meets the requirements of GDC 2 and Appendix A to 10 CFR Part 100 with respect to the capability of the structures to withstand the effects of the SSE. The design reflects, in accordance with the guidelines of SRP Section 3.7.2

- (1) consideration for the most severe earthquake appropriate for the application of the System 80+ design to the site (GDC 2) with the elimination of OBE from the design of SSC in accordance with the staff's position in SECY-93-087, as approved by the Commission on July 21, 1993 and discussed in Section 3.1.1 of this report.
- (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.
- (3) the importance of the safety functions to be performed (GDC 2) (the use of a suitable dynamic analysis or a suitable qualification test to demonstrate the SSCs can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate consideration (Appendix A, 10 CFR Part 100)).

ABB-CE has met the requirements of Item 1 (above) by using acceptable seismic design parameters in accordance with SRP Sections 3.7.1 and 2.5. The combination of earthquake loads with those resulting from normal and accident conditions in the design of seismic Category I structures as specified in SRP Sections 3.8.1 through 3.8.5, gives the assurance that the criteria under Item 2 (above) are met.

The scope of review of the seismic system and subsystem analysis for the plant included the seismic analysis methods for seismic Category I NI structures. It included review of procedures for modeling, seismic SSI, development of FRS, inclusion of torsional effects, evaluation of seismic Category I structure overturning and sliding, and determination of composite damping. The review included design criteria and procedures for evaluation of the interaction of non-seismic Category I structures with seismic Category I structures and the effects of parameter variations on FRS.

The system and subsystem analyses are performed on a linear basis. Time-history methods form the bases for the analyses of all major seismic Category I SSCs. When the modal response spectrum method is used, the methods used in combining modal responses are in conformance with the

position of RG 1.92 and also meet high-frequency mode contribution requirements. The SRSS of the maximum co-directional responses is used in accounting for three components of the earthquake motion in the response spectrum method. For the time-history analysis, three statistically independent orthogonal components of earthquake motion are applied to the structural models as separate loading cases and the total response is obtained by algebraically summing the response parameters in the time domain.

For the design of the individual members of the SSCs, the earthquake loads obtained from the seismic analysis are applied statically to an FEM and the 100-40-40 method is used in accounting for three components of the earthquake motion. In this method, 100 percent of the load from any one excitation is combined with 40 percent of the loads from the remaining two orthogonal directions using the absolute sum method. This is done for each of the three directions and the maximum load combination is used for evaluating the member adequacy. This method is equivalent to, but slightly more conservative than, the SRSS method (NUREG/CR-0098 and U.S. Nuclear Regulatory Commission, Audit summary for the March 17-18, 1993, NRC audit, Duke Engineering Services Company office, Charlotte, North Carolina). Floor spectra inputs for design and test verifications of SSCs are generated from the time-history method and they are in conformance with the position of RG 1.122. A vertical seismic system dynamic analysis is used for all SSCs where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning, sliding and uplift are considered.

A coupled structure and soil model is used to evaluate seismic SSI effects upon seismic responses. Appropriate nonlinear stress-strain and damping relationships for the soil are considered in the analysis.

3.7.2.2 Category I NNI SSCs

The staff's review of the seismic analysis of the seismic Category I NNI SSCs covers the seismic analysis methods and acceptance criteria documented in the CESSAR-DC. ABB-CE System 80+ NNI SSCs include two DFSS, two CCW Hx structures, a boric acid storage tank (BAST), buried tunnel and piping systems, and conduit banks. The DFSSs, CCW Hx structures, and pipe tunnels are reinforced concrete structures with mat foundations.

For the seismic analysis of the NNI seismic Category I structures, ABB-CE used the lumped mass model to represent the structural systems. The DFSS and CCW Hx structures were modeled, according to CESSAR-DC Sections 3.7.2 and 3.8.4, and Appendix 3.7C, Amend-

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ment U, with the masses lumped at floor levels. The lumped masses include the weight of the floor and one-half the weight of walls directly above and below, the dead weight of known equipment and components, plus 25 percent of specified live load. Each mass point has three translational and three rotational DOF. The elastic single-stick models, with beam type elements connecting the lumped masses, were developed based on the cross-sectional properties of the structural walls between the stories they represent. The stick models also accounted for the effects of both shear and flexural deformations and the torsional effects resulting from the eccentricities between the center of mass and center of rigidity of each floor. In order to ensure that the models developed will properly simulate the dynamic behavior of the structures during an earthquake event, the frequencies calculated from the fixed-base lumped mass models were tuned by the analysis of detailed FEMs of these two buildings. In addition to the torsional DOF included in the dynamic model, an eccentricity of 5 percent of the maximum building dimension, which results in an accidental torque, is applied to the static finite element structural model to calculate the element forces due to accidental torsion. Because of the axisymmetry of the building configurations, the soil-structural foundation systems of these two structures were represented by the 2-D models; however, the flexibility of the foundation mats was not considered. The techniques used for modeling the seismic Category I structures (including the consideration of accidental torsion) described above meet the guidelines of SRP Section 3.7.2 and are, therefore, acceptable. The dynamic models of these two buildings are shown in Figures 3.7C-1, 3.7C-2, 3.7C-3, and 3.7C-4 of the Appendix 3.7C to the CESSAR-DC, Amendment U.

As discussed in CESSAR-DC Section 3.7.2 and Appendix 3.7C, Amendment U, ABB-CE performed dynamic analyses of the seismic Category I NNI structures to generate the SSE responses (structural accelerations, displacements, and FRS) on a linear elastic basis. The three sets of ground motion time histories (each ground motion time history has two horizontal components and one vertical component) corresponding to the three control motions, CMS1, CMS2, and CMS3, were used as input ground motions in the seismic analyses. As discussed in Section 3.7.2.1 of this report, all three components of these ground motion time histories met both the guideline for the response spectrum enveloping and the guidelines for PSD enveloping of SRP Section 3.7.1 and are, thus, acceptable.

The seismic responses were calculated for the two horizontal directions and the vertical direction. ABB-CE did not use the technique of constant static factors for computing the vertical responses. The structural damping ratio of

7 percent used for reinforced concrete structures used complies with the SSE damping value specified in RG 1.61. The techniques used for the dynamic analyses of these structures discussed above comply with the guidelines of SRP Section 3.7.2 and are, thus, acceptable. The SSI and SSSI models of these two structures are given in CESSAR-DC Figures 3.7C-1 through 3.7C-4.

ABB-CE used the ABB-Impell version of the SASSI computer code to analyze the soil-structure system models and to generate the structural responses for accelerations, displacements, and FRS for these two buildings. As discussed in Section 3.7.2.1 of this report, this version of the SASSI computer code has been reviewed and found acceptable by the staff. Because these two buildings are located adjacent to the NI structures and the NI structures are much heavier than the NNI structures, ABB-CE considered the effects of SSSI when the structural responses in the north-south (NS) and vertical directions of the NNI structures were calculated. The procedures applied to the NNI structures seismic analysis and design, as described in CESSAR-DC Sections 3.7.2 and 3.8.4, and Appendix 3.7C, Amendment U, are as follows:

- (1) In the NS direction, using the lumped-mass NNI structural model developed above and the structural model developed for the NI structures (Section 3.7.2.1 of this report), ABB-CE developed a 2-D SASSI model with the structural embedments considered and the foundation mats assumed to be rigid. This SASSI model contained the structural model of the NI structures and one of the NNI structures. It also included the supporting soil foundation and adjacent soil.
- (2) In the east-west (EW) direction, using the lumped-mass structural model of the NNI structures, ABB-CE developed a 2-D dynamic FEM for each of the NNI structures and surrounding soil. This model did not include the SSSI effects from the NI structures.
- (3) Using the ABB-Impell version of SASSI computer code and the NS and vertical components of the artificial ground motion time histories (CMS1, CMS2 and CMS3), ABB-CE performed the SASSI analysis to generate structural responses for accelerations, displacements, and FRS in the NS and vertical directions of the DFSS and CCW Hx structures. These seismic structural responses considered the SSSI effects.
- (4) Using the SASSI computer code, the finite element soil-structure system model developed in Step 2 above and the EW components of the artificial

ground motion time histories (CMS1, CMS2 and CMS3), ABB-CE performed SSI analyses to generate seismic structural responses in the EW direction of these two structures. Because the DFSS and CCW Hx structures are axisymmetrical (rectangular shape), the torsional motion about the vertical axis need not be considered. To address the staff's concern regarding the use of only the EW component of the three ground motion time histories as input for calculating the EW structural responses, ABB-CE demonstrated, during the January 31 through February 1, 1994, audit, that the effects of the vertical component of the three ground motions on the EW structural responses are negligible.

- (5) The NS, EW, and vertical seismic structural accelerations and building displacement obtained from Steps 3 and 4 above were used as one of the design-basis loads for the structural design, and the NS, EW, and vertical FRS will be used as the input motions for the analysis of subsystems (piping systems and components) housed by these two buildings.

For the case of NNI structures founded on rock for which the SSI effects becomes negligible, a 3-D fixed-based structural model was analyzed to generate structural responses in the three directions. The analysis used computer code SAP90 and the three components of the ground motion time histories corresponding to the three design response spectra, that is, CMS1, CMS2 and CMS3, applied simultaneously. The SAP90 computer code in public domain was reviewed and validated by the staff during the previous licensing review of conventional plants such as Watts Bar. Therefore, the use of the SAP90 computer code for these two structures founded on rock is acceptable.

As a result of its review, the staff finds that the procedures used for calculating seismic responses, including FRS, of the DFSS and CCW Hx structures meet the guidelines of SRP Section 3.7.2 and are, therefore, acceptable.

As described in CESSAR-DC Section 3.7.2 and Appendix 3.7C, Amendment U, the SSI and SSSI analyses were performed to calculate the structural responses, including FRS, of the DFSS and CCW Hx structures considering all generic site conditions used in the seismic SSI analysis of the NI structures. The structural responses (structural member forces, bending moments, and displacements) corresponding to each site condition and design ground response spectrum were calculated based on the SASSI and SAP90 analyses. The final structural responses for design were calculated by enveloping all the individual responses.

For the generation of the FRS envelopes, ABB-CE (1) calculated the FRS for various damping ratios at the required locations in each of the three directions, using the 2-D finite element soil-structure system models or 3-D fixed base structural model developed for all site conditions and design ground motions considered; (2) developed the FRS envelopes from the FRS for all site conditions and design ground motions; and (3) applied a peak broadening of ± 15 percent to the FRS envelopes to account for the uncertainties associated with structural modeling, material properties, and soil dynamic moduli. On the basis of its review discussed in Section 3.7.2.1 of this report, the staff concludes that the use of the 12 generic site conditions for calculating the structural response envelopes of the DFSS and CCW Hx structures covers a wide range of site conditions and provides acceptable results for the design of the DFSS and CCW Hx structures and the subsystems housed therein. As a result of this review, the staff finds that ABB-CE's procedures for developing the structural response, including FRS envelopes meet the guidelines of SRP Section 3.7.2 and RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components," Revision 1, and are, therefore, acceptable.

In Section 5.2.4 of Appendix 3.8A to the CESSAR-DC, Amendment U, and the markups of Section 5.1.1.3 of Appendix 3.8A to the CESSAR-DC dated February 9, 1994, ABB-CE submitted the evaluation criteria and analysis procedures for evaluating dynamic stability (overturning, sliding, and flotation) of the seismic Category I structures, including the NNI structures. In addition, ABB-CE presented its calculations for staff review and demonstrated that the safety coefficients against building sliding, overturning and flotation are higher than the acceptable safety factor of 1.1 specified in SRP Section 3.8.5. As a result of its review of CESSAR-DC Appendix 3.8A, Section 5.1.1.3, Amendment V, and the calculation audit conducted, the staff concludes that ABB-CE's evaluation criteria and analysis procedures are consistent with the guidelines of SRP Section 3.7.2 and are, therefore, acceptable.

For the evaluation of the interaction of non-safety-related structures with safety-related structures, CESSAR-DC Section 3.7.2.8, Amendment U, stated that, when the safety-related structures and non-safety-related structures are integrally connected, the non-safety-related structures are analyzed and designed as a part of the safety-related structures. If these structures are adjacent to each other, in order to ensure that the failure of a non-safety-related structure under the effect of a seismic event does not impair the integrity of the adjacent safety-related structure, the evaluation procedures are as follows:

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- (1) sufficient separation between non-safety-related structures and safety-related structures is maintained, or
- (2) the non-safety-related structures are analyzed and designed to prevent their failure under SSE conditions, or
- (3) the safety-related structures are designed to withstand loads due to collapse of the adjacent non-safety-related structures if the separation criterion is not met.

The procedures for evaluating the interaction of non-safety-related structures with safety-related structures are consistent with the guidelines of SRP Section 3.7.2 and are acceptable.

The staff's evaluations of the seismic analyses and design of the specific NNI SSCs are discussed below.

Diesel Fuel Storage Structure (DFSS)

As described in CESSAR-DC Section 3.8.4 and Figure 1.2-24, Amendment U, there are two DFSSs in the System 80+ standard plant design. Each of these two buildings is a reinforced concrete box-type structure with a plan dimension of 19.2 m x 13.4 m (63 ft x 44 ft) and a height of 7.7 m (25 ft). The thickness of the walls, roof, and foundation mat is 0.7 m (2.25 ft). The embedment of the building is 3.8 m (12.75 ft) measured from the grade to the bottom of the base mat. Each of these two buildings contains two bays and each bay encloses a diesel fuel oil tank, a tank vent, a sump with a sump pump, and related piping systems.

As described in Amendment U of Appendix 3.7C to the CESSAR-DC, a dynamic model with two lumped masses connected by a massless equivalent structural member was developed to represent a DFSS in the analysis. This model coupled with the soil foundation model was used to perform the SASSI SSI and SSSI analyses or SAP90 fixed base analyses to calculate structural responses (including FRS) for each of the site conditions considered and for each set of design basis ground motion time histories. Based on the discussion above and the audit conducted from January 31 through February 1, 1994, the staff finds that the analysis procedures meet the guidelines of SRP Section 3.7.2 and the computer codes used for the analysis were accepted by the staff during previous licensing reviews. Therefore, the analysis results, including the structural response envelopes and FRS envelopes, are acceptable.

Component Cooling Water Heat Exchanger Structure

Each of the two CCW Hx structures, as described in CESSAR-DC Section 3.8.4 and Figure 1.2-25, Amendment U, is a two-story box-type reinforced concrete building with a plan dimension of 33.5 m x 13.4 m (110 ft x 44 ft) and a height of 11.6 m (38 ft). The base mat is 1.2 m (4 ft) in thickness. On the roof of the building, there are two reinforced concrete fan rooms, each located at one end of the building. The dimensions of each of these two fan rooms is 13.4 m x 7.0 m x 3 m (44 ft x 23 ft x 10 ft) and the thickness of the walls and roof is 0.7 m (2.25 ft). The embedment measured from the grade to the bottom of the base mat is 5.2 m (17 ft). One of these two buildings is located at the north side of the NI structures and the other is located at the south side. Each building houses two CCW Hxs, the CCW and station service water (SSW) piping systems, and associated valves, sumps, and sump pumps.

A three-mass lumped mass model with the massless structural members in between was developed for this building. The model coupled with the soil foundation model was also used to conduct the SASSI analyses (both SSI and SSSI analyses) or the SAP90 fixed base analyses (without soil foundation) for calculating seismic responses, including FRS, of this building. On the basis of this discussion and the audit conducted from January 31 through February 1, 1994, the staff finds that the analysis procedures meet the guidelines of SRP Section 3.7.2 and the computer code used for the analysis was accepted by the staff during previous licensing reviews. Therefore, the analysis results (including structural response envelopes and FRS envelopes) are acceptable.

Boric Acid Storage Tank (BAST)

The BAST is a flat-bottom, free-standing cylindrical steel tank anchored to the reinforced concrete foundation mat which is founded on the ground surface. In Amendment U of CESSAR-DC Section 3.7.5, ABB-CE stated that the design procedures for the seismic Category I tanks (including boric acid tank) address the guidelines documented in NUREG/CR-1161, and the criteria for the qualification of this tank addresses the design requirements and acceptance criteria for ASME Code, Section III, Class 2 and 3 components. As a result of its review, the staff finds that the analysis and design procedures described in CESSAR-DC Section 3.7.5 meet the guidelines of SRP Section 3.7.3.11.14 and are, therefore, acceptable. In addition, the design of the BAST meets design requirements and acceptance criteria for ASME Code, Section III, Class 2 and 3 components and is also acceptable.

Buried Piping, Pipe Tunnels and Conduit Banks

The design requirements and analysis procedures for the seismic Category I buried piping systems are in CESSAR-DC Section 3.7.3.12.1. The CESSAR-DC states that these piping systems are designed to meet the design requirements and acceptance criteria for ASME Code, Section III, Class 2 and 3 piping. The adequacy of these design requirements and acceptance criteria for the piping systems are discussed in Section 3.12 of this report. In this section, only the procedures that are used for the seismic analysis of buried piping are reviewed and discussed. As described in CESSAR-DC Section 3.7.3.12.1.3 and Appendix 3.8A, Amendment U, the analysis of buried piping considers the strains (axial and bending) and associated stresses from the effects of seismic wave passage, seismic differential movements between the soil and anchor points, and thermal expansion and contraction. In addition, the effects of the lateral soil pressure and the groundwater effects acting on the piping are also considered in the design. These analysis procedures meet the guidelines of SRP Section 3.7.3.11.12 and are, thus, acceptable.

For the seismic Category I pipe tunnels, an equivalent static analysis was performed. In the analysis, the strains (axial and bending) and associated stresses due to the seismic wave passage and seismic differential movements between the soil and anchor points were considered. In addition, the lateral dynamic soil pressure due to earthquake was also taken into consideration. As stated in CESSAR-DC Appendix 3.8A, Section 11.8, the same analysis approach is applied for the analysis of cable tunnels and conduit banks. These analysis procedures meet the guidelines of the SRP Section 3.7.3 and are, thus, acceptable.

3.7.3 Seismic Subsystem Analysis

The scope of the staff's review of the seismic subsystem analyses included the seismic analysis methods for CE System 80+ seismic Category I SSCs that are not explicitly contained in the structural models for the seismic analyses of seismic Category I structures. Such items include all seismic Category I cable trays and supports, conduit and supports, HVAC duct and supports, above-ground tanks and structural elements that support other seismic Category I items, the dynamic effects of which could affect the seismic response of the supported items. Seismic analysis of piping and pipe supports, seismic qualification of seismic Category I mechanical equipment and seismic qualification of seismic Category I instrumentation and electrical equipment is evaluated in Sections 3.9 and 3.10 of this report.

CESSAR-DC Section 3.7.3 states that seismic Category I structures, subsystems, and components are analyzed by the response spectrum or time-history method as described in CESSAR-DC Section 3.7.2.1.1 or by an equivalent static method as described in CESSAR-DC Section 3.7.3.5. These analysis methods are in accordance with the guidelines of SRP Section 3.7.2, Item II.1, referenced by SRP Section 3.7.3, Item II.1 and, therefore, are acceptable.

CESSAR-DC Section 3.7.3.3 supplies information on modeling techniques. The procedure for dynamic decoupling of subsystems and for modeling criteria, respectively, are in accordance with the guidelines of SRP Section 3.7.2, Items II.3.b and Item II.1.a.(iii). The use of these criteria for the purposes stated are also in accordance with SRP Section 3.7.3, Items II.1 and II.3 guidelines, and are acceptable.

CESSAR-DC Section 3.7.3.4 states that the basis for the acceptability of subsystems is that the stresses and deformations produced by the vibratory motion of the postulated seismic events combined with other design loadings are within the established limits. In addition, CESSAR-DC Section 3.7.3.4, provides information on the design approach to avoid resonance of components and subsystems. In this approach, within practical limitations, the fundamental frequencies of the components and subsystems are designed to be "sufficiently removed" from the resonant range. If resonance does occur, the resonance should be accounted for in the analysis. This procedure is in compliance with the SRP Section 3.7.3 guideline that equipment in the resonance range is adequately designed for the applicable loads and is acceptable.

CESSAR-DC Sections 3.7.2.15 and 3.7.3.15 provide information on procedures for damping for the modal superposition and the direct integration methods, respectively. CESSAR-DC Section 3.7.3.6 and Table 3.7-1 give additional information on damping. These procedures are in accordance with RGs 1.61 and 1.84 and the SRP Section 3.7.3, Item II.5, criteria and are acceptable.

CESSAR-DC Section 3.7.3.6 (which references CESSAR-DC Section 3.7.2.6) states that for the time-history method of analysis, three statistically independent earthquake motions (2 horizontal and 1 vertical) are used as input and the total response is obtained by algebraically summing the responses in the time domain. For the response spectrum method, the total response due to the three independent components of earthquake motion (2 horizontal and 1 vertical) are obtained by combining the separate responses by the SRSS method. This is in accordance with the guideline of SRP Section 3.7.3, Item II.6, and is acceptable.

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CESSAR-DC Section 3.7.3.7 states that for the response spectrum method of analysis (1) the SRSS method of combination will be used for modes that are not spaced closely and (2) the RG 1.92 method of combination will be used for closely spaced modes. These provisions are in accordance with the guidelines of SRP Section 3.7.3, Item II.7, and are acceptable.

CESSAR-DC Section 3.7.3.11 states that torsional effects due to eccentric masses are considered for subsystems including piping. This is in compliance with SRP Section 3.7.3, Item II.11, and is acceptable.

Before the issuance of the DSER, ABB-CE stated in CESSAR-DC Section 3.7.3.12 that the intake structure will be designed so that "differential movement between this structure and the earth is negligible." The staff noted that ABB-CE gave no indication of the magnitude of this permissible motion or the level of stresses associated with such "negligible" relative motion. By DSER Open Item 3.7.3-7, the staff noted that ABB-CE should provide generic approaches for the evaluation of the intake structure as well as acceptance criteria that will be used to evaluate that structure. Subsequently, in various meetings with the staff, ABB-CE stated that the design approaches and acceptance criteria for evaluating intake structure is a site-specific item. CESSAR-DC Table 1.9-1, Amendment U, included this item. This is acceptable and resolves DSER Open Item 3.7.3-7.

On the basis of its review, the staff concludes that the seismic subsystem analysis procedures and criteria proposed by ABB-CE provide an acceptable basis for the seismic design of subsystem and components.

The staff concludes that the seismic subsystem analysis procedure and criteria are acceptable and meet the requirements of GDC 2 and 10 CFR Part 100, Appendix A. This conclusion is based on ABB-CE's meeting the requirements of GDC 2 and 10 CFR Part 100, Appendix A with respect to the capability of the structural subsystems to withstand the effects of the earthquakes so that their design reflects the following:

- (1) consideration for the most severe earthquake appropriate for the application of the System 80+ design to the site (GDC 2) with the elimination of OBE from the design of SSC in accordance with the staff's position in SECY-93-087, as approved by the commission on July 21, 1993 and discussed in Section 3.1.1 of this report
- (2) appropriate combination of the effects of normal and accident conditions with the effect of the natural phenomenon
- (3) the importance of the safety functions to be performed (GDC 2); the use of a suitable dynamic analysis or a suitable qualification test to demonstrate that SSCs can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate consideration (10 CFR Part 100, Appendix A)

ABB-CE meets the requirements of Item 1 (listed above) by use of the acceptable seismic design parameters as per SRP Section 3.7.1. The combination of loads resulting from earthquakes with those resulting from normal and accident conditions in the design of seismic Category I structural subsystems is in conformance with Item 2 (listed above).

3.7.3.1 Ductwork, Cable Tray, Conduits, and Supports

In the DSER, the staff identified Open Item 3.9.3-9 which stated that the applicant should provide explicit information about the duct support design and reconcile the commitments for the duct and duct support design with the EPRI Utility Requirements Document.

The details of the HVAC ductwork analysis and design procedures are given in CESSAR-DC Appendix 3.9A, Section 2. Seismic Category I HVAC ductwork is designed and supported to withstand the loads and load combinations presented in CESSAR-DC Appendix 3.9A, Sections 2.2 and 2.3, respectively. The analysis and design guidelines ensure that the HVAC ductwork and supports will be within the allowable stress and deflection criteria under the design loads and load combinations. In areas where non-safety-related ductwork passes over or near safety-related equipment or components, the supports/restraints and duct system is designed using seismic Category I criteria to prevent any damage, degradation, or interference with the performance of equipment required for safety functions.

The design loads for seismic Category I ductwork and supports are specified in CESSAR-DC Appendix 3.9A, Section 2.2, which include internal pressure, dead and live load, thermal effects, seismic loads, and wind and tornado loads, if applicable. The design load combinations specified in CESSAR-DC Appendix 3.9A, Section 2.3, include normal, severe, extreme, and abnormal load combinations. The design loads and load combinations are in accordance with the guidelines of SRP Section 3.8.4 and, therefore, acceptable. The analysis procedures and acceptance criteria for seismic Category I ductwork and supports are specified in CESSAR-DC Appendix 3.9A, Section 2.4. The damping values used are in accordance with RG 1.61 and therefore, acceptable. The effect of

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eccentricity of forces relative to the duct centerline is considered. The seismic analysis of seismic Category I ductwork and supports is performed using the static coefficient method, response spectrum modal analysis or the time-history analysis. These methods are applied in accordance with the guidelines of SRP Section 3.7.3 and, therefore, are acceptable. The allowable stress criteria for seismic Category I ductwork and supports are specified in CESSAR-DC Appendix 3.9A, Section 2.5. The allowable stress criteria are established using conservative values in compliance with the requirements of American Institute of Steel Construction (AISC) "Manual of Steel Construction, Allowable Stress Design" and ANSI/AISC 690-84, as discussed in Section 3.8.4.1 of this report, and are acceptable.

The details of the cable tray/conduit and supports analysis and design procedures are in CESSAR-DC Appendix 3.9A, Section 3. Seismic Category I cable tray/conduit and supports are designed and supported to withstand the loads and load combinations presented in CESSAR-DC Appendix 3.9A, Section 3.2 and 3.3, respectively. The analysis and design guidelines ensure that the cable tray/conduit and supports will be within the allowable stress and deflection criteria under the design loads and load combinations. In areas where non-safety-related cable tray and/or conduit passes over or near safety-related equipment or components, the tray, conduit, and support/restraint systems are design using seismic Category I criteria to prevent any damage, degradation, or interference with the performance of the safety-related degradation equipment.

The design loads for seismic Category I cable tray/conduit and supports are specified in CESSAR-DC Appendix 3.9A, Section 3.2, which include dead and live loads, thermal effects, and seismic loads. The design load combinations are specified in CESSAR-DC Appendix 3.9A, Section 3.3, which include normal and extreme load combinations. The design loads and load combinations are in accordance with the guidelines of SRP Section 3.8.4 and are acceptable. The damping values are specified in CESSAR-DC Appendix 3.9A, Section 3.4 and are in accordance with RG 1.61 and are acceptable. The seismic analysis procedure for seismic Category I cable tray/conduit and supports is specified in CESSAR-DC Appendix 3.9A, Section 3.5. The seismic analysis of seismic Category I cable tray/conduit and supports is performed using the static coefficient method, response spectrum modal analysis or the time history analysis. These methods are applied in accordance with the guidelines of SRP Section 3.7.3 and, therefore, are acceptable. The allowable stress criteria for seismic Category I cable tray/conduit and supports is specified in CESSAR-DC Appendix 3.9A, Sections 3.3 and 3.5.4. The allowable

stress criteria are established using conservative values in compliance with the requirements of AISI standard for carbon steel and stainless steel cold-formed sections (American Iron and Steel Institute, "Cold-Formed Steel Design Manual," 1986, Washington, D.C.) and ANSI/AISC 690-84 for structural steel members, bolts and welds, as discussed in Section 3.8.4.1 of this report, and are acceptable.

The staff concludes that the design of seismic Category I ductwork and supports, cable tray and supports, and conduit and supports is acceptable and meets the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, and 4. This conclusion is based on the following:

- (1) ABB-CE has met the requirements of 10 CFR 50.55a and GDC 1 with respect to assuring that the safety-related ductwork and supports and cable tray/conduit and supports are designed, fabricated, constructed, tested and inspected to quality standards commensurate with their safety function by meeting the guidelines of RGs and industry standards indicated below.
- (2) ABB-CE has met the requirements of 10 CFR Part 50, Appendix A, GDC 2 by designing the safety-related ductwork and supports and cable tray/conduit and supports to withstand the most severe earthquake that has been established with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loading such as earthquake and other natural phenomenon.
- (3) ABB-CE has met the requirements of 10 CFR Part 50, Appendix A, GDC 4 by assuring that the design of the safety-related ductwork and supports and cable tray/conduit and supports are capable of withstanding the effects of the conditions associated with NOP and postulated accidents.

The criteria used in the analysis, design, and construction of safety-related ductwork and supports and cable tray/conduit and supports to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff. The use of these criteria as defined by the applicable codes, standards, and specifications, the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; and the materials and the testing give reasonable assurance that, in the event of winds, tornados, earthquakes, and various postulated accidents, the safety-related

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ductwork and supports and cable tray/conduit and supports will withstand the specified design conditions without impairment of structural integrity or the performance of required safety function. On this basis, DSER Open Item 3.9.3.1-9 is resolved.

3.7.4 Seismic Instrumentation

As described below, the seismic instrumentation system specified for the System 80+ design in CESSAR-DC Section 3.7.4 is acceptable and meets GDC 2 and 10 CFR Part 100, Appendix A. ABB-CE meets these requirements by requiring installation of instrumentation that is capable of adequately measuring the effects of an earthquake.

As specified in CESSAR-DC Section 3.7.4.1, five triaxial accelerometers will be installed: one on the common basemat of the NI, one in the shield building, one at the operating floor of the containment structure, one on the foundation mat of the CCW Hx building, and one at El. 28 m (91.75 ft) of the CCW Hx building; also, a free field instrument will be installed at the finished grade. The requirements specified in CESSAR-DC Sections 3.7.4.3 and 3.7.4.5 constitute an acceptable program for obtaining data on seismic ground motion as well as data on the amplitude, frequency, and phases relationship of the seismic response of major structures and systems in the event of an earthquake. A readout of pertinent data from the various seismic instruments will yield sufficient information to guide the operator on a timely basis to determine if the level of earthquake ground motion requiring plant shutdown has been exceeded. Data obtained from such installed seismic instrumentation will be sufficient to ascertain that the seismic analysis assumptions and the analytical models used in the seismic design of the System 80+ were adequate and that allowable stresses have not been exceeded under conditions when continuity of operation is intended. The staff finds the design for seismic instrumentation acceptable.

In view of the time and effort required to determine if the level of earthquake ground motion requiring plant shutdown has been exceeded, COL applicants should establish plant operating procedures that define specifically what constitutes a significant exceedance of the level of earthquake ground motion requiring plant shutdown. This was DSER Open Item 3.7.4-1. CESSAR-DC Section 3.7.4.4 includes this information and Sections 3.7 and 3.7.4.6 state that the COL applicant will submit to the NRC, as part of its application, procedures for pre-earthquake planning and post-earthquake actions. On this basis, DSER Open Item 3.7.4-1 is resolved. This is included in COL Action Item 3.7-1.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

This section is not applicable to the ABB-CE System 80+ design because a concrete containment is not used.

3.8.2 Steel Containment

Section 3.8.2.1 of the CESSAR-DC describes the ABB-CE System 80+ SCV and the containment penetrations, including the equipment hatch, personnel locks, fuel transfer penetration, and the mechanical and electrical penetrations. The containment is a spherical, welded steel vessel structure designed in accordance with Section III, Division 1, Subsection NE, Class MC components of the ASME Code, 1989 Edition. The diameter of the containment is 61 m (200 ft) and the plate nominal thickness is 43.8 mm (1.75 in.). The containment plate material is ASME A537 Class 2. In the transition region, where the containment enters the concrete (El. 28 m (91 ft-9 in.)), the containment shell thickness is 50 mm (2 in.). Above elevation 28 m (91 ft-9 in.), the containment is an independent, free-standing structure. Below this elevation, the vessel is encased between the base slab of the ISs and the shield building foundation. The embedded portion of the steel containment shell is connected to the shield building foundation and the base slab of ISs by radially extending shear transfer plates to resist sliding due to the effects of earthquakes.

The equipment hatch, personnel locks, and the fuel transfer penetration are designed, fabricated, and tested in accordance with Section III, Subsection NE of the ASME Code. The equipment hatch consists of a cylindrical sleeve in the SCV shell and a dished head 6.7 m (22 ft) in diameter with mating bolted flanges. The two personnel locks are 3.05 m (10 ft) in diameter and each has double doors with an interlocking system to prevent both doors from being opened simultaneously. Seals for the equipment hatch and the personnel locks are designed to maintain containment integrity under design-basis accident conditions, including pressure, temperature, and radiation. The containment performance under severe accident conditions is discussed in Chapter 19 of this report.

Mechanical penetrations are treated as fabricated piping assemblies and meet the requirements of the ASME Code, Section III, Subsections NE and NC. Mechanical bellows are supplied where necessary to minimize reactions on the SCV. The bellows are designed to maintain containment integrity under all design load conditions.

The containment electrical penetration assemblies are designed, fabricated, and tested in accordance with

IEEE-317," Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations." The pressure boundary portion of the assembly is designed, fabricated, and tested in accordance with Section III, Subsection NE of the ASME Code. The containment electrical penetration assemblies, including seals, are designed to maintain containment integrity under design-basis accident conditions, including pressure, temperature, and radiation.

Before issuing the DSER, the staff raised a question (Q220.41, letter from T.V. Wambach (NRC) to E.H. Kennedy (ABB-CE), September 26, 1991) about the stability (i.e., possibility of overturning or sliding) of the steel containment and the ISs during an earthquake. ABB-CE's calculations (Letter from C.B. Brinkman (ABB-CE) to NRC, LD-92-0166, February 12, 1992) determined that both structures will be stable, with no relative movement of structures, for design-basis loading cases in combination with SSE. In the DSER, the staff had expressed the concern that ABB-CE had not evaluated earthquakes beyond the SSE, such as the review level earthquake (RLE) of approximately twice the magnitude of SSE as stated in SECY-93-087; this evaluation is needed for the seismic margins assessment (SMA). The staff's position on design for beyond-design-basis-event scenarios (NRC, SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," S.J. Chilk, to J.M. Taylor, June 26, 1990) is that the stability problem associated with beyond-design-basis earthquakes be addressed in the probabilistic risk assessment (PRA)-based seismic margin evaluation as part of the design certification for the System 80+ plant.

ABB-CE presented the results of its evaluation of the potential for the containment sliding and overturning due to the design-basis earthquake and the RLE for seismic margins in the July 28, 1993 audit meeting (audit summary for the July 28, 1993, NRC audit, Rockville, Maryland) with the staff, and summarized it in CESSAR-DC Section 3.8.2.4.3. ABB-CE considered four potential failure modes: overturning of the IS independent of the SCV, overturning of the IS together with the SCV, sliding between the IS and the SCV, and sliding between the IS and the SCV combined with respect to the supporting concrete. The stability requirements defined by ABB-CE are (1) the line of action of the reaction force is located below the base of the crane wall and (2) shear bars are designed to resist the full sliding forces without consideration for frictional resistance for the design-basis earthquake and considering friction for the RLE. The stability analysis is performed in the time domain. The earthquake levels applied are a RLE of 0.6g, used for the determination of seismic margin, applied at the free ground surface,

and the design-basis earthquake (SSE) of 0.3g, as specified in the CESSAR-DC. The acceleration time histories for the "worst case" soil conditions are determined by comparing the acceleration profiles of the IS for the three control motions CMS1, CMS2, and CMS3 for the various soil conditions specified in the CESSAR-DC. On the basis of the results of the analysis and the technical discussions during the meeting, ABB-CE committed that positive shear transfer mechanism will be provided to connect the SCV to the shield building foundation slab and the base slab of the ISs. Subsequently, ABB-CE incorporated the design basis and details of this shear transfer mechanism into CESSAR-DC Section 3.8.2.4.3, Amendment U. The factor of safety against sliding and overturning of the SCV is 2.4, which is acceptable.

At the transition region between the free-standing part and the encased portion of the steel containment, compressible material is provided to reduce bearing loads on the concrete as well as to reduce thermally induced local bending stresses in the steel containment. In the DSER, the staff had expressed its concern (Q220.42 and Q220.43) about the mechanical properties of this compressible material and the stress conditions and buckling potential of the steel containment in this region because ABB-CE did not present information in the CESSAR-DC concerning: (1) composition of the compressible material, (2) how these material properties are obtained, (3) the uncertainty these material properties have, and (4) accessibility for periodic inspection. ABB-CE stated (LD-92-0166, February 12, 1992) that the material is self-expanding cork and its static mechanical properties are determined from a load deflection test. ABB-CE responded in the meeting (presentation material from the April 27 through 30, 1992, NRC/ABB meeting, Bethesda, Maryland) that the transition region will be available for periodic inspection over the lifetime of the plant. ABB-CE also stated that the staff could examine the design information related to the transition region. In the DSER, the staff stated that ABB-CE's responses are acceptable, except that ABB-CE should also address the issues associated with (1) the uncertainty of the mechanical properties of and the EQ as well as age-related degradation management for a 60-year design life for this compressible material (DSER Open Item 3.8.2-1) and (2) the measure(s) to be implemented to prevent collection of moisture in the transition region (DSER Open Item 3.8.2-2). By DSER Open Item 3.8.2-12, the staff also stated that ABB-CE should provide a corrosion analysis of the SCV for a 60-year plant design life.

To address these open items, ABB-CE performed a stress analysis of the SCV to take into account the variation in the material properties of the compressible material. As a result, ABB-CE revised the CESSAR-DC to describe the steps to be taken to inspect the compressible material and

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to prevent corrosion of the SCV during the life of the plant. These steps include coating the SCV near the transition region, sealing concrete to preclude moisture intrusion, visual inspection of the containment base metal and welds in accordance with ASME Code Section XI and 10 CFR Part 50, Appendix J, use of sloped floors and drains to prevent collection of moisture in the transition region, provisions for the removability of the compressible material, and a design to preclude equipment or ductwork from being located such that it inhibits a visual inspection at the steel concrete interface. Furthermore, the SCV thickness of 50 mm (2 in.) in the transition region is beyond design requirements and allows for corrosion over the 60-year life of the plant. The staff reviewed these provisions and determined they were adequate to address the concern of the degradation of the material properties of the compressible material over the 60-year life of the plant and the corrosion potential due to the collection of moisture in the transition region. On this basis, DSER Open Items 3.8.2-1, 3.8.2-2, and 3.8.2-12 are resolved.

Before issuing the DSER, the staff requested (Q220.49, letter from T.V. Wambach (NRC) to E.H. Kennedy (ABB-CE), September 26, 1991) that information on taking measurements at critical locations during preoperational structural integrity testing of the steel containment be provided because this information may be useful in validating the containment analysis methods. Subsequently, ABB-CE stated (LD-92-0166, February 12, 1992) that the preoperational structural integrity testing data will be taken as described in CESSAR-DC Section 14.2.12.1.130, "Containment Integrated Leak Rate Test and Structural Integrity Test." ABB-CE's response with respect to the type, amount and location of data collection is adequate and acceptable.

In RAI Q220.50, the staff expressed concern about the potential sources of a missile or high-energy-line break against the steel containment. ABB-CE stated that there are no high-energy lines between the steel containment and the operating floor and refueling cavity walls. However, there are high-energy lines between the secondary shield wall and the steel containment, and between the steel containment and the shield building. ABB-CE stated that high-energy lines, if not eliminated from break potential by LBB analysis, will be kept from being a source of hazard to safety-related components and equipment by separation, guard pipes, shields, and whip restraints. Between the steel containment and the shield building, high-energy lines will be enclosed in guard pipes eliminating any potential missiles. Postulated missiles from equipment in the containment are listed in CESSAR-DC Table 3.5-1, and the missile protection requirements are given in CESSAR-DC Section 3.5.1. ABB-CE's response satisfactorily addressed the staff's concern.

On containment shell stress analysis, in SRP Section 3.8.2, the staff recommends that the stresses in the containment for the load combinations of SRP Table 3.8-2 satisfy the limits prescribed in SRP Table 3.8.2-1. Before issuing the DSER, ABB-CE performed a 3-D shell analysis of the System 80+ containment for the loading combinations specified above. Based on vendor data, the compressible material in the transition region was taken to have a modulus of 488 kPa/cm (180 psi/in.). The FEM included the equipment hatch and personnel air locks. Two mesh sizes were used. In the DSER, the staff noted that ABB-CE would satisfy the SRP guidelines subject to resolution of DSER Open Items 3.8.2-4 and 3.8.2-5. DSER Open Item 3.8.2-4 required that ABB-CE submit for staff review the stress analysis results for the most highly stressed meridian, as presented at the April 29, 1992, meeting (Ref. 14) between the NRC staff and ABB-CE representatives. DSER Open Item 3.8.2-5 required that ABB-CE describe the method that it plans to use to verify that the designs of the penetrations and penetration reinforcements satisfy the stress limits of SRP Table 3.8.2-1.

On containment shell buckling, in SRP Section 3.8.2, the staff recommends that the stresses in the containment satisfy the stability requirements of ASME Code Subsection NE and RG 1.57 for loading combinations that include compressive stresses. Before issuing the DSER, ABB-CE performed an elastic bifurcation buckling analysis of the System 80+ containment for the compressive loading combination. The 3-D shell FEM model that was used included the equipment hatch and the personnel airlock penetrations. The compressible material in the transition region had a modulus of 488 kPa/cm (180 psi/in.). For a comparison of the finite element results to the classical solution, a capacity reduction factor of 0.125 was applied to the closed-form buckling stress to account for the differences between classical shell buckling theory and experimental buckling results applicable to shells with imperfections. In the DSER, the staff noted that ABB-CE would satisfy these criteria subject to resolution of DSER Open Items 3.8.2-6, 3.8.2-7, 3.8.2-8, and 3.8.2-9.

DSER Open Item 3.8.2-6 required ABB-CE to verify that the mesh size of the finite element is sufficiently small to achieve convergence of the ANSYS bifurcation buckling load. DSER Open Item 3.8.2-7 required that ABB-CE substantiate the buckled shape resulting from the previous analysis or perform an additional analysis to eliminate the anomalies. DSER Open Item 3.8.2-8 noted that ABB-CE should justify the acceptability of a factor of safety of 2 for stability with Level C loading condition. DSER Open Item 3.8.2-9 requested that ABB-CE submit for staff review the pre-buckling stresses for the most highly

stressed meridian and verify that stresses at buckling are within the elastic range.

Subsequently, ABB-CE presented the results of the SCV stress and buckling analysis in an audit meeting with the staff on August 10 and 11, 1993 (audit summary for the August 10 and 11, 1993, NRC audit at the ABB-CE office in Windsor, Connecticut), and later summarized in the CESSAR-DC, Amendment U. Major issues discussed by ABB-CE in its presentation and summarized in the CESSAR-DC include containment technical data and configuration, SCV analysis for design basis loads, 3-D FEM of the SCV, description of design loadings including seismic input spectra, loading combinations, material properties, buckling analysis methods, modeling of imperfections, compliance with ASME Code allowables for service loads and stability safety factors for buckling analysis and large deflection analysis.

The service load stress analysis for Level A through D conditions is performed using the ANSYS Version 4.4(a), a 3-D, finite-element code based on linear-elastic, thin-shell theory. The analysis is performed for the design loads and load combinations described in SRP Section 3.8.2 and CESSAR-DC Section 3.8.2.4.2 and Table 3.8.2. In the audit (audit summary for the August 10-11, 1993, NRC audit at the ABB-CE office in Windsor, Connecticut), ABB-CE also presented the allowable stress intensities specified in SRP Section 3.8.2 and CESSAR-DC Table 3.8-3E for the primary general membrane stresses (P_m), primary local membrane stresses (P_L), local membrane plus bending stresses ($P_L + P_b$), and local membrane and bending plus secondary stresses ($P_L + P_b + Q$) in accordance with the ASME Code Section III, Subsection NE. ABB-CE presented the results for the maximum element stress intensities for the test, design, and the service level conditions in CESSAR-DC Table 3.8-3E, Amendment U, which demonstrates that the stress intensities are within the allowable limits of the ASME Code. On this basis, DSER Open Item 3.8.2-4 is resolved.

In accordance with CESSAR-DC Section 3.8.2.1.1, the SCV will be thickened around the penetrations to compensate for the loss of area due to openings using the area replacement rule of ASME Code Section III, Subsection NE, Paragraph NE-3332. The additional thickness will depend upon the nominal size, thickness and location of the penetration sleeve and will be in accordance with the ASME Code. This clarification is acceptable and resolves DSER Open Item 3.8.2-5.

In the August 10 and 11, 1993, audit, ABB-CE also presented the methodology for its SCV buckling analysis using the ANSYS finite element code. The methodology

has been summarized in the CESSAR-DC (Amendment U). ABB-CE used the ASME Code Section III, Subsection NE, Paragraph 3222 rigorous analysis methodology which considers the effect of gross and local buckling, geometric imperfections, nonlinearities, large deflections, and inertial forces. ABB-CE performed the rigorous analysis by using the large deflection elastic-plastic option in an ANSYS static analysis. With the large deflection option, the deflection under load is continuously used to redefine the geometry of the structure, thus producing a revised stiffness matrix. By observing the rate of change in deflection per iteration, an estimate of the stability of the structure can be made. If the rate of change is increasing, the loading is above critical and is an indication of buckling instability. If the rate of change is decreasing or constant, the structure is at or below the critical buckling load. The total load levels at the last converged load step are divided by the nominal load levels to determine the safety factor.

In order to verify the validity of the ANSYS buckling analysis results and to demonstrate that the finite element mesh size is sufficiently small to have achieved convergence of the ANSYS buckling load, ABB-CE performed the stability analysis of a 3-D full sphere and demonstrated that the results are comparable to the classical full-sphere theoretical buckling solution.

In the August 10-11, 1993, audit, ABB-CE also presented the results of the SCV buckling analysis using the ANSYS computer code for a 3-D containment FEM. These results are summarized in CESSAR-DC Amendment U. The analysis is done taking into account an external design-basis pressure of 13.8 kPa (2 psi), gravity loads due to the weight of the SCV, and seismic loads applied as nodal forces based on nodal accelerations and masses. The direction and combination of force resultants are varied to determine the critical direction of application. The material properties of the SCV are based on the operating temperature of 43.3 °C (110 °F). The stress-strain curve is modified to account for potential residual stresses by applying a plasticity reduction factor taken from ASME Code Case N-284. The imperfections are modeled based on the ASME Code, Article NE-4221, requirement that the maximum deviation from true theoretical form for a vessel of the System 80+ dimensions designed for external pressure shall not exceed the thickness of the vessel. The imperfection is assumed to be axisymmetric. For the Service Level A stability analysis, ABB-CE applied the design-basis accident pressure of 365 kPa (53 psi) and design-basis accident temperature of 143 °C (290 °F) in addition to the dead and live loads. The analysis is performed with an imperfection of a full sine wave with a half-wavelength of 2.4 m (8 ft) and a peak-to-peak ampli-

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tude of 43.8 mm (1.75 in.). The SCV stability safety factor for Level A loads is 3.0.

For the Service Level C and D stability analyses, ABB-CE applied the external pressure of 13.8 kPa (2 psi) due to inadvertent containment spray actuation and SSE loads in addition to the dead and live loads. The Service Level C and D stability analyses are completed for two types of imperfections: (1) a full sine wave with a half-wavelength of 2.4 m (8 ft) and a peak-to-peak amplitude of 43.8 mm (1.75 in.) and (2) a half sine wave with a half-wavelength of 4.9 m (16 ft) and a peak-to-peak amplitude of 43.8 mm (1.75 in.). The resulting SCV Service Level C stability factors for these two types of imperfections are 2.70 and 2.74, respectively and exceed the ASME Code, NE-3222, required factor of safety of 2.5.

In CESSAR-DC Section 3.8.2.4.3, Amendment U, ABB-CE presented the results for the ultimate load analysis of the containment. The ultimate capacity is determined using an elastic analysis with the axisymmetric model. The internal pressure is increased until the maximum stress intensity reaches the Service Level C allowable membrane stress intensity, which equals the nominal yield stress value for the given temperature. Temperature values of 65.6 °C (150 °F), 143.3 °C (290 °F), 176.7 °C (350 °F), and 232.2 °C (450 °F) are evaluated. As summarized in Table 3.8-3D of the CESSAR-DC, the minimum ultimate pressure capacity at 232 °C (450 °F) is 910 kPa (132 psi). This satisfies the requirements of GDC 50 as related to steel containment being designed with sufficient margin of safety to accommodate appropriate design loads.

In CESSAR-DC Section 3.8.2.4.3, Amendment U, ABB-CE also presented the evaluation for the combustible gas loading using an elastic analysis with the axisymmetric model. The peak pressure from hydrogen combustion and the design-basis pressure are added together and applied as an internal pressure to the SCV. The maximum calculated stress intensity is within the Level C allowable stress intensity limit as summarized in CESSAR-DC Table 3.8-3E, Amendment U.

In addition to the stress and buckling analyses done by ABB-CE for the SCV, the staff's consultants at Ames National Laboratory have also performed an independent review of the structural design of the ABB-CE System 80+ SCV. The results of the Ames work are documented in NUREG/CR 5957 (U.S. NRC, "System 80+ Containment Structural Design Review," 1993) and in NUREG/CR 6161 (U.S. NRC, "Buckling Evaluation of System 80+ Containment," June 1994). The stress analysis of the SCV is performed using the BOSOR4 finite difference computer program. The evaluation of the SCV against buckling is performed by using BOSOR5 finite

difference and ABAQUS finite element computer codes. For the stress analysis, the containment is modeled as an axisymmetric shell consisting of different segments and mesh points with the additional mass of the penetrations and appurtenances being smeared around the circumference. The transition region near the base is modeled using elastic springs with a foundation modulus of 488 kPa/cm (180 psi/in.). The structural analysis of the containment shell is performed for each individual load case using the nonlinear elastic axisymmetric stress analysis option of BOSOR4. The results are combined to calculate the stress intensities for the design conditions and Service Levels A, B, C, and D, which are compared to allowable stress intensities from the ASME Code. The results confirm ABB-CE's conclusion that the calculated stress intensities are below the ASME Code allowable limits.

The buckling analysis of System 80+ SCV was done by Ames using the computer code BOSOR5 for the axisymmetric case and ABAQUS for the 3-D case, taking into account the non-axisymmetric seismic loads as well as the effect of equipment hatch and personnel locks. For the axisymmetric case, the loading combination for the buckling evaluation is chosen on the basis of the stress analysis as that loading which resulted in the largest compressive stress. The controlling load case for Level C is the inadvertent spray actuation resulting in external pressure of 13.8 kPa (2 psi) and a temperature of 43.3 °C (110 °F) in addition to the seismic loading due to SSE. The buckling analysis is performed on the basis of the worst meridian assumption, that is, the maximum stresses are assumed to exist uniformly around the circumference. The material nonlinearities and residual stresses are incorporated using a stress-strain constitutive relationship derived from the ASME Code Case N-284. For the System 80+, the perfect shell with an elastic material was initially analyzed. A sinusoidal imperfection with a radial amplitude of 21.9 mm (0.875 in.), which corresponds to the ASME Code specified maximum deviation of one shell thickness of 43.8 mm (1.75 in.), is introduced and the imperfection wavelength is varied to identify the critical imperfection configuration. Sensitivity studies are conducted to investigate the effects of mesh size and transition zone stiffness on the controlling buckling load. The factor of safety against buckling is defined as a load multiplier which is equal to the ratio of the loads at which buckling occurs to the input loads. For the axisymmetric case using BOSOR5, the calculated factor of safety for Level C is 2.35. This is less than the factor of safety of 2.5 prescribed by Section NE-3222 of the ASME Code but more than the factor of safety of 1.67 specified by ASME Code Case N-284 or the factor of safety of 2 prescribed by RG 1.57. Since the load is applied in a conservative manner as discussed above, and the factor of safety is acceptable according to ASME Code Case N-284 and

RG 1.57, the staff concludes that the factor of safety for buckling of 2.35 for Level C is acceptable.

As a prelude to the non-axisymmetric buckling analysis of the System 80+ SCV, Ames verified the ability of the computer code ABAQUS to predict the critical load for elastic buckling problems. Ames analyzed a circular cylinder under axial load and a spherical shell under uniform radial pressure, and demonstrated that the results are in agreement with the classical theoretical solution. In addition, Ames formulated and analyzed three sample studies with the computer codes ABAQUS and BOSOR using a nonlinear stress-strain relationship, and compared the results in order to optimize the finite element type and mesh size, and to benchmark the computer program ABAQUS. The sample studies considered were System 80+ SCV buckling analysis with uniform external pressure, System 80+ SCV buckling analysis due to its own weight, and buckling of a reinforced opening with a ring load. Subsequent to the program verification, the System 80+ SCV 3-D FEM was formulated, including the equipment hatch, two personnel locks, and the embedded region. An imperfection sensitivity study was performed in order to determine the imperfection to be modeled in the System 80+ SCV FEM. The first several vibration modes were calculated, SSE response spectrum was entered, and the SRSS accelerations were converted to nodal inertial forces for application in the buckling analysis. One hundred percent of the seismic inertial forces were applied in the direction of the equipment hatch barrel, with 40 percent in the other two orthogonal directions. As a result of the buckling analysis for the Level C loading, the load multiplier was determined to be 2.4, compared to the factor of safety 2.35 from the axisymmetric buckling analysis using BOSOR5 and the factor of 2.7 determined by ABB-CE using ANSYS. Three other loading conditions were investigated with full seismic loading in (1) a direction opposite to the equipment hatch barrel, (2) the vertical direction, and (3) the direction of the lower personnel air lock.

On the basis of the results of the buckling evaluations performed by ABB-CE and independent verification analyses performed by the staff's consultant at Ames, the staff concludes that the System 80+ SCV has factors of safety against buckling that meet the ASME Code and is acceptable. This resolves DSER Open Items 3.8.2-6, 3.8.2-7, 3.8.2-8, and 3.8.2-9.

In addition to the analysis for design-basis loading, Ames also performed a buckling analysis for the System 80+ containment under seismic loading beyond SSE. The objective of the analysis was to evaluate the containment buckling potential for seismic loading at least twice the SSE and inclusive of such sustained loads as dead weight.

In this analysis, the axisymmetric model was used and the material model consisted of the effective stress-strain curve for the SCV material, SA537 Class 2, including residual stress effects. The factor of safety against buckling, λ , is defined as a load multiplier equivalent to the ratio between the seismic load which causes buckling, to the SSE loads. The unfactored dead load is added to the factored SSE loads. Hence, the load on the structure is $D + \lambda$ (SSE). In the analysis, λ is increased until buckling occurs. The minimum predicted buckling load is determined by introducing the geometric imperfections and the material nonlinearities into the analysis and performing an imperfection sensitivity analysis by varying the imperfection wavelength parameter. The minimum buckling load multiplier was determined to be 2.9, indicating that the System 80+ SCV is not susceptible to buckling unless subjected to a seismic loading as high as approximately 3 times the SSE.

By DSER Open Item 3.8.2-3, staff noted that ABB-CE should address the containment shell seismic fragility and the containment performance in its PRA evaluation for the beyond-design-basis events. ABB-CE presented its PRA-based seismic-margins analysis methodology and results in CESSAR-DC Section 19.7.5. The staff's evaluation of the System 80+ seismic margin assessment is included in Section 19.7.5 of this report. This resolves DSER Open Item 3.8.2-3.

ABB-CE submitted an ultimate capacity analysis of the containment. The current staff position on containment performance under severe accident conditions is delineated in SECY-93-087 and the corresponding SRM (NRC, Staff requirement memorandum (SRM), SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (LWR) Designs," S.J. Chilk to J.M. Taylor, July 21, 1993). By DSER Open Item 3.8.2-10, the staff requested that ABB-CE verify that the Sandia strain criteria (NRC, NUREG/CP-0095, "Evaluation of the Performance of the Sequoyah Unit 1 Containment Under Conditions of Severe Accident Loading," Paper No. SAND88-1631C, pp 571-588, 1988) have been satisfied for all strains in the axisymmetric analysis model. By DSER Open Item 3.8.2-11, the staff requested that ABB-CE describe the method to be used to verify that the strains at all discontinuities (i.e., around penetrations and penetration reinforcements) satisfy the Sandia strain criteria. ABB-CE's presentation of the severe accident phenomenology and containment performance for the System 80+ is documented in Section 19.11 of the CESSAR-DC. The staff's safety evaluation of the System 80+ containment performance under severe accidents is discussed in Section 19.11 of this report. This resolves DSER Open Items 3.8.2-10 and 3.8.2-11.

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In order to reconcile the as-built data with the structural design-basis loads, a structural analysis report will be prepared for the SCV as required by ASME Code, Section III. This report will document that the containment structure meets the requirements specified in CESSAR-DC Section 3.8.2 and that design changes and identified construction deviations which could potentially affect the structural capability of the containment have been incorporated into the structural analysis. Records for construction material properties, as-built dimensions and arrangements, and design documents for the containment structure will be reviewed. Deviations from the design will be considered acceptable provided an evaluation is performed to verify that the structural design meets the requirements of CESSAR-DC Section 3.8.2. The as-built structural analysis report will be prepared by the COL applicant. This is included in COL Action Item 3.8-1.

The material, quality control, and construction techniques for the SCV are in CESSAR-DC Section 3.8.2.6. The SCV material, fabrication, erection, and nondestructive examination are in accordance with Subsection NE, Class MC components of the ASME Code, Section III, 1989 Edition. The general QA provisions for the SCV are as outlined in Chapter 17 of the CESSAR-DC, supplemented by the ASME Code. The SCV is ASME Code stamped and, therefore, the ASME Code requirements for quality control have priority over those outlined in Chapter 17 of CESSAR-DC in case of any conflict.

Construction techniques for the assembly of the SCV are described in CESSAR-DC Section 3.8.2.6.3. Testing and in-service surveillance requirements for the SCV are in CESSAR-DC Section 3.8.2.7. The SCV, personnel airlocks, and equipment hatch are inspected and tested in accordance with the ASME Code Section III, Subsection NE, and the penetrations are pressure tested as prescribed in Subsection NC of the ASME Code. In accordance with CESSAR-DC Section 3.8.2.7, periodic leakage rate tests of the SCV are conducted in accordance with 10 CFR Part 50, Appendix J.

The staff's review finds that sufficient information is given in accordance with the guidelines of SRP Section 3.8.2. The staff concludes that the design of the steel containment is acceptable and meets the relevant requirements of GDC 1, 2, 4, 16, 50, and 1. The conclusion is based on the following:

- (1) ABB-CE meets the recommendations of SRP Section 3.8.2 and the requirements of GDC 1 by ensuring that the steel containment is designed, fabricated, erected, constructed, tested, and inspected to the quality standards commensurate with its

safety function to be performed by meeting the guidelines of RGs and industry standards.

- (2) ABB-CE meets the requirements of GDC 2 by designing the steel containment to withstand a severe earthquake that is more severe than the SSEs for all currently licensed plants (with the exception of the two coastal California sites) with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loading such as earthquakes and other natural phenomena.
- (3) ABB-CE meets the requirements of GDC 4 by designing the steel containment to withstand the effects associated with system and equipment failures, including missiles, pipe whip, and loads associated with a spectrum of LOCAs.
- (4) ABB-CE meets the requirements of GDC 16 by designing the containment so that it is an essentially leaktight barrier to prevent the uncontrolled release of radioactive effluent to the environment.
- (5) ABB-CE meets the requirements of GDC 50 by designing the steel containment to accommodate, with sufficient margin, the design leakage rate and the calculated pressure and temperature conditions resulting from accident conditions, and by ensuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, ABB-CE has used the recommendations of RGs and industry standards. ABB-CE has also performed appropriate analysis to demonstrate that the ultimate capacity of the containment will not be exceeded and has established the minimum margin of safety for the design.
- (6) The ABB-CE System 80+ primary containment is a welded steel vessel fabricated to the requirements of ASME Code, Section III. The ASME Code requires that the vessel materials meet the fracture toughness requirements of Subsection NE-2000. The staff concludes the design is in compliance with the requirements of GDC 51 because the steel vessel is made of materials that will meet the fracture toughness requirements of the ASME Code. This will ensure that the SCV materials will not undergo brittle fracture and the probability of a rapidly propagating fracture will be minimized.

The criteria used in the analysis, design, and construction of the steel containment structure to account for anticipated loadings and postulated conditions that may be imposed on

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the structure during its service lifetime conform with established criteria, codes, standards, and guides and the industry standard, ASME Code, Section III, Division 1, Subsection NE, 1989 Edition.

Any change to the use of ASME Code, Section III, Division I, Subsection NE, 1989 Edition for the design and construction of the System 80+ SCV would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation. Any requested change to the use of this code shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

The use of these criteria as defined by the applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials and quality control programs; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within and outside the plant, the structure will withstand the specified conditions without impairment of structural integrity or of the performance of required safety functions. Furthermore, the seismic Category I concrete shield building protects the steel containment from the effects of wind and tornados and various postulated accidents occurring outside the shield building.

3.8.3 Concrete and Structural Steel Internal Structures

CESSAR-DC Section 3.8.3 describes the internal structure and its design requirements. The IS is a group of reinforced concrete structures that enclose and support the systems and components inside the steel containment. These structures also serve as biological shields and as protection from missiles. The concrete base of the IS rests inside the lower portion of the SCV and is connected to it with radially extending shear bars. With the exception of platforms and miscellaneous steel, the ISs are seismic Category I. The major ISs are the primary shield wall, the secondary shield wall, the refueling cavity, the in-containment refueling water storage tank (IRWST), various floors including the operating floor, and the support systems for the reactor vessel, steam generators, RCPs, and primary loop piping.

The primary shield wall, which encloses the reactor vessel and protects the vessel from internal missiles and serves as a biological shield, is a minimum of 1.8 m (6 ft) thick.

The secondary shield wall (crane wall) supports the polar crane, protects the SCV from internal missiles, and serves as biological shield for the coolant loop and equipment.

The crane wall is a right cylinder with an inside diameter of 39.6 m (130 ft), a height of 36 m (118 ft), and a minimum thickness of 1.2 m (4 ft).

The refueling cavity comprises the storage area for the upper guide structure, the storage area for the core support barrel, and the refueling cavity. The refueling cavity, when filled with borated water, forms a pool above the reactor vessel. The shield walls that form the refueling cavity are a minimum of 1.8 m (6 ft) thick.

The IRWST is a single source of water for the SI and containment spray pumps and a heat sink for the safety depressurization system (SDS). The IRWST has a stainless steel liner to prevent leakage, is dish-like in shape, and utilizes the lower section of the IS as its boundary. A complete description of the IRWST and its design basis loadings is in CESSAR-DC Section 6.8.

The operating floor provides access for operating personnel and serves as a biological shield. The operating floor consists of a reinforced concrete slab inside the crane wall and a steel grating outside the crane wall.

The support systems for the reactor vessel, steam generators, RCPs and primary loop piping are described in CESSAR-DC Section 5.4.14.

CESSAR-DC Section 3.8.3 and Appendix 3.8A describe the applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, structural acceptance criteria, material and quality control procedures, and testing and in-service surveillance requirements for the ISs which are in compliance with the guidelines of SRP Section 3.8.3.

Concrete and structural steel ISs are designed to resist dead and live loads, operating thermal and pipe reaction loads, hydrostatic and earth pressure loads, accident temperature and pressure loads, and seismic loads.

Design load combinations for concrete and structural steel ISs include normal operating loads, normal operating loads combined with extreme environmental loads, normal operating loads combined with abnormal loads, and normal operating loads combined with abnormal and extreme environmental loads. These load combinations are described in CESSAR-DC Appendix 3.8A, Section 5.0. The load combinations described in CESSAR-DC Appendix 3.8A, Section 5.0, for seismic Category I structures are in compliance with the guidelines of SRP Section 3.8.3 and are, therefore, acceptable.

The IS and the supporting substructure are modeled with 3-D solid, plate or shell and beam finite elements in order

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to determine the design forces and moments from the applied loads and load combinations.

During the NRC and ABB-CE meeting on April 27 to April 30, 1992, the staff audited the methodology for considering the effects of concrete cracking in the seismic analysis of ABB-CE System 80+ structures. SRP Section 3.7.2 specifies that consideration should be given in the analysis, to the effects on FRS, of expected variations of structural properties. Prior to the issuance of the DSER, ABB-CE stated that concrete cracking was not considered explicitly in the seismic analysis of Category I structures. ABB-CE contended that it was implicitly considered through the SSI analyses of the 13 soil cases discussed in CESSAR-DC Section 2.5 (ABB-CE, "System 80+ Standard Design - CESSAR-DC Design Certification," Amendment I, Windsor, Connecticut, December 21, 1990). These analyses cover a wide range of soil column frequencies to account for structural frequency changes as related to cracked concrete cross-sections. This answer (the implicit consideration of concrete cracking) was not technically acceptable to the staff and the staff stated in the DSER that further justification was required. The staff also asked that the effects of concrete cracking be addressed for all seismic Category I structures. Subsequently, ABB-CE performed a parametric study to assess the effects of cracked concrete on the in-structure response spectra and presented the results in CESSAR-DC Appendix 3.7B. ABB-CE performed an SSI analysis assuming that the concrete has cracked during the SSE event and that the stiffness of all concrete structural elements was reduced to 70 percent of its original (uncracked) value. For concrete structures, such as the IS, it is shown that the effects of concrete cracking caused peak shifting of less than 15 percent. The increase in amplitude near the predominant spectral peak was below the envelope of all the soil cases. ABB-CE concluded that since several different cases are considered in the SSI analyses, the frequency shift and amplitude change due to concrete cracking is covered by the 15 percent spectra broadening criteria established for the design of piping, components and equipment. For the SCV, concrete cracking made almost no difference in its response. On the basis of these studies, the staff concludes that the effects of concrete cracking have been adequately addressed and the guidelines of SRP Section 3.7.2 for the procedure to consider the effects of the expected variations of structural properties such as concrete cracking on the seismic analysis of seismic Category I structures are satisfied. This resolves DSER Open Item 3.8.3-1.

Structural acceptance criteria for the concrete and structural steel ISs are described in CESSAR-DC Section 3.8.3.5 and Appendix 3.8A, Section 5.0. The allowable limits are

in conformance with the guidelines of SRP Section 3.8.3 and are, therefore, acceptable.

CESSAR-DC Tables 3.8-4 and 1.8-6 list ACI 349-85 and ANSI/AISC N690-84 as the standards for the design of concrete and steel structures, respectively. ACI 318-89, Chapter 21, is used to determine the detailing requirements at structural joints and regions where reinforcing bars are spliced for seismic Category I structures. The use of ACI 318-89, Chapter 21, for detailing requirements amounts to an enhancement to the requirement of complying with ACI 349, as specified in SRP Section 3.8.3, and is, therefore, acceptable to the staff. The staff provides its positions on ACI 349-85 and ANSI/AISC N690-84 to ABB-CE and appends them to this chapter in Appendices 3A and 3B. ABB-CE incorporated these supplementary provisions in CESSAR-DC Section 3.8.4.5. The staff's evaluation of ABB-CE's compliance with ACI 349-85 and ANSI/AISC N690-84 is discussed in Section 3.8.4 of this report. Any change to the use of ANSI/AISC Standard N690 (1984 Edition) and ACI Standard 349 (1985 Edition) for the design and construction of containment internal structural elements would involve an unreviewed safety question and, therefore, require NRC review and approval before implementation. Any requested change to the use of these codes shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

In the October 28 and 29, 1993, audit (audit summary for the October 28 and 29, 1993, NRC audit at the Duke Engineering Services Company office in Charlotte, North Carolina), the staff audited several selected areas in the NI structures on the structural design and detailing considerations, including the ISs. The areas were selected primarily from seismic loading considerations, based on ABB-CE's experience in designing several nuclear power facilities. The main purpose of the audit was to review and discuss the design calculations for these areas, evaluate the design adequacy, and obtain insights of the design process for the purpose of incorporation into the CESSAR-DC. In order to provide ductility, which would prevent brittle or catastrophic failure, in seismic Category I internal reinforced concrete structures, Chapter 21 of ACI 318 is used to determine the required anchorage and splicing of connections, the configuration of reinforcing steel in the structural joints, the regions where reinforcing bars are spliced, and the required placement of stirrups and hoop steel. CESSAR-DC Appendix 3.8B describes design and analysis results for the selected areas of the ISs. A detailed review of the analysis and design of the ISs for the selected areas provides the staff confidence that the guidelines of SRP Section 3.8.3 are complied with and the overall design is acceptable and that the critical dimensions and member sizes are adequate.

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The design of ISs addresses the vertical alignment of the crane wall with the corresponding wall below the SCV. A 125-mm (5-in.) vertical misalignment of the center line of the crane wall above and below the SCV is evaluated, and the resulting stresses are found acceptable for the typical reinforcing pattern determined for the crane wall. The design also considers potential differential basemat settlement and the effect on the crane wall alignment.

Materials, quality control, and special construction techniques for the concrete ISs are as described in CESSAR-DC Section 3.8.4 and the staff evaluation is in Section 3.8.4 of this report.

On the basis of its review, the staff determines that sufficient information is provided in the CESSAR-DC in accordance with the guidelines of SRP Section 3.8.3. The staff concludes that the design of the containment ISs is acceptable and meets the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, 4, 5, and 50. The conclusion is based on the following:

- (1) ABB-CE meets the recommendations of SRP Section 3.8.3 and the requirements of 10 CFR 50.55a and GDC 1 by ensuring that the containment ISs are designed, fabricated, erected, constructed, tested, and inspected to the quality standards commensurate with its safety function to be performed by meeting the guidelines of RGs and industry standards.
- (2) ABB-CE meets the requirements of GDC 2 by designing the containment ISs to withstand a severe earthquake that is more severe than the SSEs for all currently licensed plants (with the exception of the two coastal California sites) with sufficient margin, and the combinations of the effects of normal and accident conditions with the effects of environmental loading such as earthquakes and other natural phenomena.
- (3) ABB-CE meets the requirements of GDC 4 by designing the containment ISs to withstand the effects associated with system and equipment failures, including missiles, pipe whip, and loads associated with a spectrum of LOCAs.
- (4) ABB-CE meets the requirements of GDC 5 by demonstrating that the structural systems and components are not shared between units.
- (5) ABB-CE meets the requirements of GDC 50 by designing the containment ISs to accommodate, with sufficient margin, the design leakage rate and the calculated pressure and temperature conditions

resulting from accident conditions, and by ensuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, ABB-CE has used the recommendations of RGs and industry standards. ABB-CE has also performed appropriate analysis to demonstrate that the ultimate capacity of the structures will not be exceeded and has established the minimum margin of safety for the design.

The criteria used in the analysis, design, and construction of the containment ISs to account for anticipated loadings and postulated conditions that may be imposed on the structures during their service lifetime conform with established criteria, codes, standards, and RGs acceptable to the staff. These include meeting the positions of RGs 1.94 and 1.142 and industry standards ACI 349-85 and ANSI/AISC N690-84.

The use of these criteria as defined by the applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials and quality control programs; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified conditions without impairment of structural integrity or of the performance of required safety functions.

3.8.4 Other Category I Structures

3.8.4.1 RB and NA Structures

CESSAR-DC Section 3.8.4 describes seismic Category I structures other than the steel containment and the IS and their design requirements. The description for the RB is provided in CESSAR-DC Section 3.8.4.1.1 and the NA structures are described in CESSAR-DC Section 3.8.4.1.2.

The RB comprises the containment shield building and subsphere in addition to the SCV and the ISs. With the exception of platforms and miscellaneous steel, these structures are seismic Category I.

The containment shield building is a reinforced concrete structure composed of a right cylinder with a hemispherical dome; it shares a common foundation mat with the NA structures. The containment shield building serves as a biological shield and as protection from external missiles for the SCV and safety-related equipment enclosed in the containment building. The containment shield building has an inner radius of 32 m (105 ft) and a wall thickness of 1.2 m (4 ft) up to El. 44.5 m (146 ft). Above El. 44.5 m

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(146 ft), and including the dome, the shield building wall thickness is 0.9 m (3 ft). The height of the containment shield building is approximately 65.5 m (215 ft). The structural outline of the containment shield building is shown in CESSAR-DC Figures 1.2-2 and 1.2-3. Above the concrete base, there is an annular space between the SCV and the containment shield building for structural separation. The shield building and the NA are connected to form a monolithic structure.

The subsphere is the portion of the RB which is below EL. 28 m (91 ft 9 in.), external to the SCV and internal to the containment shield building and houses auxiliary safety-related equipment.

The NA comprises the control complex, diesel generator areas, MSVH areas, CVCS and maintenance areas, spent fuel storage area, and the unit vent. The NA, with the exception of the unit vent, is a seismic Category I reinforced concrete structure comprising walls, columns, beams, and floor slabs. The NA shares common walls and foundation basemat with, and is monolithically connected to, the RB, forming the NI. Structural details of the nuclear system complex are shown in CESSAR-DC Figures 1.2-2 through 1.2-10.

CESSAR-DC Section 3.8.4 in conjunction with Appendix 3.8A provide the applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, structural acceptance criteria, material and quality control procedures, and testing and in-service surveillance requirements for the RB and the NA structures.

The RB and the NA structures are designed to resist dead and live loads, operating thermal and pipe reaction loads, hydrostatic and earth pressure loads, accident temperature and pressure loads, and seismic loads.

Before issuing the DSER, the staff asked, in RAI Q220.51 (letter from T.V. Wambach (NRC) to E.H. Kennedy (ABB-CE), September 26, 1991), if the dynamic effects of pressure loads due to pipe breaks are considered in the design. ABB-CE stated (LD-92-016, February 12, 1992) that the dynamic effect of pressure loads for postulated pipe breaks that are not eliminated by LBB evaluations is considered for design-basis conditions. The staff interpreted this response as a commitment by ABB-CE to design all subcompartments for global pressure/temperature effects, unless otherwise justified. In the DSER, the staff noted that, based on the staff evaluation of "LBB Design Basis" in Section 3.6.3 of the DSER, ABB-CE's response is acceptable subject to resolution of DSER Open Item 3.8.4-2 which stated that ABB-CE should clarify, in

the CESSAR-DC, the commitment to design all subcompartments for global pressure/temperature effects. CESSAR-DC Amendment R provided the subcompartment pressure and temperature for the MSVH but did not provide similar information for the pipe chase containing CVCS and other subcompartments. Subsequently, ABB-CE revised CESSAR-DC Section 3.8.4.5 to state that the structural analysis report, as required by Tier 1 certified design material, will document that the structural design has been verified for subcompartment pressurization and anchor loads. This is acceptable and resolves DSER Open item 3.8.4-2.

Design load combinations for the RB and the NA structures include normal operating loads, normal operating loads combined with extreme environmental loads, normal operating loads combined with abnormal loads, and normal operating loads combined with abnormal and extreme environmental loads. These load combinations are described in CESSAR-DC Appendix 3.8A, Section 5.0. The load combinations described in CESSAR-DC Appendix 3.8A, Section 5.0, for seismic Category I structures are in compliance with the guidelines of SRP Section 3.8.4 and are, therefore, acceptable.

The containment shield building is analyzed with a 3-D FEM to determine the forces and moments due to the applied loads and load combinations.

The NI, consisting of the RB and NA structures, is designed to prevent overturning, sliding, and flotation. ABB-CE has completed the calculations to verify the overturning and sliding stability of the NI.

The nuclear system complex is analyzed with a detailed 3-D FEM, using shell, plate, and beam elements, for the design load and load combinations to determine the distribution of the loads to all structural components. Out-of-plane design forces and moments are determined by manual calculations or local area models. The details of the models and methods of application of loads including seismic effects are in CESSAR-DC Appendix 3.8A.

Structural acceptance criteria for the RB and the NA structures are described in CESSAR-DC Section 3.8.4.5 and Appendix 3.8A, Section 5.0. The allowable limits are in conformance with the guidelines of SRP Section 3.8.4 and are, therefore, acceptable.

CESSAR-DC Tables 3.8-4 and 1.8-6 list ACI 349-85 and ANSI/AISC N690-84 as the standards that will be used for design of concrete and steel structures, respectively. ACI 318-89, Chapter 21, is used to determine the detailing requirements at structural joints and regions where reinforcing bars are spliced for seismic Category I structures.

The use of ACI 318-89, Chapter 21, for detailing requirements is acceptable. The use of the ACI 349-85 code requirements for the design of the seismic Category I reinforced concrete structures is acceptable, except for the design requirements for the steel embedments; the staff position on these requirements should be satisfied. The staff position on steel embedment design is described in Appendix 3A to this chapter. CESSAR-DC Section 3.8.4.5 incorporates the staff position as a part of design criteria. This is acceptable. CESSAR-DC Section 3.8.4.5 describes the limitations for the use of ANSI/AISC N690-84 in the design of steel structures. The use of the ANSI/AISC N690-84 standard with these limitations complies with the staff position described in Appendix 3B to this chapter. Any change to the use of ANSI/AISC Standard N690 (1984 Edition) and ACI Standard 349 (1985 Edition) for the design and construction of RB and NA structural elements would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Any requested change to the use of these codes shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

In the October 28-29, 1993, audit (audit summary for the October 28-29, 1993, NRC audit at the Duke Engineering Services Company office in Charlotte, North Carolina), the staff also audited several selected areas in the RB and the NA structures including structural design and detailing considerations. The areas were selected primarily from seismic loading considerations and ABB-CE's experience in designing several nuclear power facilities. The main purpose of the audit was to review and discuss the design calculations for these areas, evaluate the design adequacy, and obtain some insights of the design process for the purpose of incorporation into the CESSAR-DC.

FEMs are used for global in-plane distribution of forces and the out-of-plane design calculations are performed based on more detailed local models or hand calculations with proper assumptions on end conditions and load distribution. In order to provide ductility in the seismic Category I RB and the reinforced concrete NA structures, Chapter 21 of ACI 318 is used to determine the required anchorage and splicing of connections, the configuration of reinforcing steel in the structural joints, the regions where reinforcing bars are spliced, and the required placement of stirrups and hoop steel. CESSAR Appendix 3.8B provides design and analysis results for the selected areas of the RB and the NA structures. A detailed review of the analysis and design of the RB and the NA structures for the selected areas gives the staff confidence that the guidelines of SRP Section 3.8.4 are complied with and overall design is acceptable and that the critical dimensions and member sizes are adequate.

Major materials used in the construction of the RB and the NA structures are concrete, reinforcing bars, and structural steel. Cement for concrete will be Type I or II conforming to "Standard Specification for Portland Cement," American Society for Testing and Materials (ASTM C150). Aggregates for concrete will conform to "Standard Specification for Concrete Aggregate," ASTM C33. Water used in mixing concrete will be clean and free from injurious amounts of oil, acids, alkalis, salts, organic materials or other substances that may be deleterious to concrete or steel. The mixing water properties will be compared with distilled water by performing the tests described in CESSAR-DC Section 3.8.4.6.1. Admixtures, if used, will conform with the applicable ASTM standard in CESSAR-DC Section 3.8.4.6.1. In order to prevent corrosion of reinforcing bars, the combined chloride content of the admixtures and mixing water will not exceed 250 ppm. The ingredient materials will be stored in accordance with the recommendations of ACI 304 and the concrete mixes will be designed in accordance with ACI 301. Reinforcing steel will consist of deformed reinforcing bars conforming to ASTM A615, Grade 60 or ASTM A706, Grade 60. The fabrication and fabrication tolerances of reinforcing bars will be in accordance with CRSI MSP-1, "Manual of Standard Practice." The placement of reinforcing bars, including spacing of bars, concrete protection of reinforcement, splicing of bars, and field tolerances will be in accordance with ACI 349-85. Epoxy coated steel is used wherever a corrosive environment is encountered. The structural steel will consist of low carbon steel conforming to ASTM A36 or other structural steels listed in ANSI/AISC N690-84. Fabrication and erection of structural steel will be in accordance with the requirements of ANSI/AISC N690-84. The welded structural connections will be in accordance with the requirements of ANSI/AISC N690-84, and bolted connections will be made with high strength bolts conforming to either ASTM A325 or A490. The quality control of materials will be in accordance with the relevant ASTM specifications and the overall QA program described in CESSAR-DC Chapter 17 supplemented by the special provisions of ACI 349-85, ASTM A615 or A706, and ANSI/AISC N690-84. Cylindrical and dome portions of the shield building will be constructed using standard construction techniques and testing and surveillance during construction will be in accordance with ACI 349-85, ACI 301, ANSI/AISC 690-84 or ANSI N45.2.5, as applicable. Compliance with industry standards indicated above provides assurance that seismic Category I structures are fabricated, erected, and constructed to quality standards commensurate with their safety functions.

In order to reconcile the as-built data with the structural design-basis loads, a structural analysis report will be prepared for seismic Category I structures. This report

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will document that the structures meet the requirements specified in CESSAR-DC Section 3.8 and that design changes and identified construction deviations which could potentially affect the structural capability have been incorporated into the structural analysis. Records for construction material properties for concrete, reinforcing steel and structural steel, as-built dimensions and arrangements, and design documents for the structures will be reviewed. Deviations from the design will be considered acceptable if an evaluation is performed to verify that the structural design meets the requirements of CESSAR-DC Section 3.8 and if the seismic FRS of the as-built structure do not exceed the design-basis FRS by more than 10 percent.

On the basis of its review, the staff determines that sufficient information is provided in the CESSAR-DC in accordance with the guidelines of SRP Section 3.8.4. The staff concludes that the design of the RB and the NA structures is acceptable and meets the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, 4, and 5. The conclusion is based on the following:

- (1) ABB-CE meets the recommendations of SRP Section 3.8.4 and the requirements of 10 CFR 50.55a and GDC 1 by ensuring that the RB and the NA structures are designed, fabricated, erected, constructed, tested, and inspected to the quality standards commensurate with its safety function to be performed by meeting the guidelines of RGs and industry standards.
- (2) ABB-CE meets the requirements of GDC 2 by designing the RB and the NA structures to withstand a severe earthquake that is more severe than the SSEs for all currently licensed plants (with the exception of the two coastal California sites) with sufficient margin, and the combinations of the effects of normal and accident conditions with the effects of environmental loading such as earthquakes and other natural phenomena.
- (3) ABB-CE meets the requirements of GDC 4 by designing the RB and the NA structures to withstand the effects associated with system and equipment failures, including missiles, pipe whip, and loads associated with a spectrum of LOCA.
- (4) ABB-CE meets the requirements of GDC 5 by demonstrating that the structural systems and components are not shared between units.
- (5) ABB-CE meets the requirements of Appendix B because its QA program provides adequate mea-

sures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of the RB and the NA structures to account for anticipated loadings and postulated conditions that may be imposed on the structures during their service lifetime conform with established criteria, codes, standards, and RGs acceptable to the staff. These include meeting the positions of RGs 1.69, 1.94, 1.115, and 1.142 and industry standards ACI 349-85 and ANSI/AISC N690-84.

The use of these criteria as defined by the applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials and quality control programs; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the structures, the RB and the NA structures will withstand the specified conditions without impairment of structural integrity or of the performance of required safety functions.

3.8.4.2 Category I NNI Structures

Four seismic Category I NNI structures fall into the scope of the CE System 80+ standard design: two DFSS and two CCW Hx structures. As described in CESSAR-DC Subsection 3.8.4.1, Amendment U, the DFSS and CCW Hx structures are box-type reinforced concrete structures with mat foundations. One DFSS and one CCW Hx structure are located at the north side of the NI structures and the others are located at the south side. The DFSS has a 19.2 m x 13.4 m (63 ft x 44 ft) plan dimension and is 7.6 m (25 ft) in height; the plan dimension of the CCW Hx structure is 33.5 m x 13.4 m (110 ft x 44 ft) and the height is 11.6 m (38 ft). All the walls, roofs, floors, and foundation mats are 0.061 m (2 ft 3 in.) thick, except for the CCW Hx structure foundation mat which is 1.2 m (4 ft) thick and the flat slab which is 0.92 m (3 ft) thick.

In CESSAR-DC Section 7.1 of Appendix 3.8A, Amendment U, ABB-CE stated that a static 3-D FEM was developed for the DFSS. Computer code ANSYS was used to analyze this structure for the combined load conditions of dead load, live loads, tornado loads (including missiles), temperature loads, and SSE seismic loads. These combined load conditions were modeled as static loads in the ANSYS model. The ANSYS computer code is in the public domain and has been reviewed and validated by the staff during a previous licensing review. Therefore, the use of ANSYS code is acceptable. In addition to the design loads stated above, as described in Section 5.0 of CESSAR-DC Appendix 3.8A, live loads

due to precipitation (rain, snow, and ice), lateral soil pressure due to the soil density and the effects of ground water, hydrostatic loads associated with ground water and exterior flood water, and wind loads were included in the design. These loads for the DFSS design are summarized as follows:

- maximum tornado wind speed 531.1 km/hr (330 mph)
- tornado missiles In accordance with SRP Section 3.5.1.4 Spectrum II, Region I
- live loads due to precipitation 2.39 kPa (50 psf)
- design wind speed 196.5 km/hr (122.1 mph)

When the design loads and combined load conditions were modeled, the three orthogonal components of earthquake loads (two horizontal and one vertical) were considered statically and simultaneously applied on the structures. On top of these seismic loads, an additional eccentricity of ± 5 percent of the maximum building dimensions at the level under consideration was assumed to account for accidental torsion. The other design loads were also applied statically and directly on the structures.

On the basis of its review of CESSAR-DC and the audit conducted on January 31 through February 1, 1994, the staff concludes that the approach of considering the seismic loads (including dynamic soil pressure due to earthquake) and other design loads for the structural design is acceptable.

The ANSYS analysis results (structural member forces, shear forces and bending moments) form the design basis for the DFSS. As described in CESSAR-DC Section 3.8.4.4 and Appendix 3.8A, major materials used in the design and construction of the DFSS are concrete, reinforcing bars, and structural steel. Cement for concrete will be of Type I or II conforming to "Standard Specification for Portland Cement," ASTM C150. Aggregates for concrete will conform to "Standard Specification for Concrete Aggregate," ASTM C33. Water used in mixing concrete will be clean and free from injurious amounts of oil, acids, alkalis, salts, organic materials or other substances that may be deleterious to concrete or steel. The proposed mixing water properties will be compared with distilled water by performing the tests described in CESSAR-DC Section 3.8.4.6.1. Admixtures, if used, will conform with the applicable ASTM standard in CESSAR-DC Section 3.8.4.6.1. In order to prevent corrosion of reinforcing bars, the combined chloride

content of the admixtures and mixing water will not exceed 250 ppm. The ingredient materials will be stored in accordance with the recommendations of ACI 301 and the concrete mixes will be designed in accordance with ACI 301. Reinforcing steel will consist of deformed reinforcing bars conforming to ASTM A615, Grade 60 or ASTM A706, Grade 60. The fabrication and fabrication tolerances of reinforcing bars will be in accordance with CRSI MSP-1, "Manual of Standard Practice." The placement of reinforcing bars, including spacing of bars, concrete protection of reinforcement, splicing of bars and field tolerances will be in accordance with ACI 349. Epoxy coated reinforcing steel bars are used wherever a corrosive environment is encountered. For calculating the development length, CESSAR-DC Appendix 3.8A, Section 6.2.1.1.1, states that the required splice length given in ACI 349 Section 12.2.2 shall be increased using the factors provided in ACI 318 Section 12.2.4.3. The structural steel will consist of low carbon steel conforming to ASTM A36 or other structural steels listed in ANSI/AISC N690. Fabrication and erection of structural steel will be in accordance with the requirements of ANSI/AISC N690-84. The welded structural connections will be in accordance with the requirements of ANSI/AISC N690-84, and bolted connections will be made with high strength bolts conforming to either ASTM A325 or A490. The quality control of materials will be in accordance with the relevant ASTM specifications and the overall QA program described in CESSAR-DC Chapter 17 as supplemented by the special provisions of ACI 349, ASTM A615 or A706, and ANSI/AISC N690. The strength of the construction materials, f'_c for the compressive strength of concrete and f_y the yield stress of steel, for the DFSS are as follows:

- $f'_c = 27.6 \text{ MPa (4,000 psi)}$ for concrete
- $f_y = 413.7 \text{ MPa (60,000 psi)}$ for reinforcing steel
- $f_y = 248.2 \text{ MPa (36,000 psi)}$ for structural steel

For the design of the CCW Hx structure, as described in CESSAR-DC Section 3.8.4, the same design-basis loads and combined load conditions and the same approach for modeling the loads as used for the DFSS design were considered. The construction materials and specification of the materials are the same as those used for the DFSS. Instead of modeling the structure by a 3-D FEM and using the ANSYS computer code for the analysis, formulas based on the theory of beams and plates were used and hand calculations were performed for computing the structural member forces for the design. On the basis of its review of the CESSAR-DC and the audit conducted on January 31 and - February 1, 1994, the staff concludes that the procedures for computing the structural member forces

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of the CCW Hx structure meet the guidelines of SRP Section 3.8.4 and the computer code used was previously accepted by the staff. Therefore, the calculated structural member forces are acceptable.

The analysis and design of the reinforced concrete portion of the structures used the ultimate strength design method in accordance with ACI 349. The seismic Category I structural steel analysis and design were in accordance with ANSI/AISC Standard N-690. CESSAR-DC Section 3.8.4.4 also describes the limitation of using ACI 349 code and ANSI/AISC N-690 standard for the design of seismic Category I structures. The use of these two design codes with the limitations complies with the staff positions in Appendices 3A and 3B to this chapter.

During the audit on January 31 and February 1, 1994, the staff raised a concern that the CESSAR-DC should provide the building joint design details for reinforcing steels in the CESSAR-DC. Subsequently, ABB-CE provided the staff with these details in the markups for the locations and details to be provided in future CESSAR-DC Amendments. This was part of Confirmatory Item 3.8.4.2-1 of the advance version of the SER. In Amendment V, ABB-CE provided these details in CESSAR-DC Figures 3.8B-5 through 3.8B-9, which are acceptable. This resolves the applicable part of Confirmatory Item 3.8.4.2-1.

On the basis of its review of the CESSAR-DC and the design calculation audit conducted on January 31 - February 1, 1994, the staff concludes that the design of the DFSS and CCW Hx structures is acceptable.

In order to ensure that the as-built DFSS and CCW Hx structures are able to withstand the structural design-basis loads described above, ABB-CE, in CESSAR-DC Section 3.8.4.5.3, states that a structural analysis report will be prepared for seismic Category I structures. This report will document that the structures meet the design requirements specified in CESSAR-DC Section 3.8, and design changes and identified construction deviations, which could potentially affect the structural capability of the structure, have been incorporated into the structural analysis. Specifically, the following records will be reviewed:

- construction records stating material properties for concrete, reinforcing steel, and structural steel
- as-built structural dimensions and arrangements
- design documents for the structures

This structural analysis report will summarize the results of the reviews, evaluations and corrective actions. Devia-

tions and design changes from the original design are acceptable provided the following acceptance criteria are met:

- An evaluation is performed.
- The structural design meets the requirements specified in CESSAR-DC Section 3.8.4.
- The FRS of the as-built structure does not exceed the design-basis FRS by more than 10 percent.

On the basis of the preceding discussion, the staff concludes that the procedures for the reconciliation analysis will ensure that the as-built DFSS and CCW Hx structures are able to withstand the structural design-basis loads and combined load conditions defined in CESSAR-DC Section 3.8.4 and are, thus, acceptable.

As described in Sections 11.7 and 11.8 of Appendix 3.8A to the CESSAR-DC, Amendment U, the CCW tunnel, buried cable tunnels, and conduit banks are classified as seismic Category I underground structures. The procedures and the design loads and load combinations for the analysis and design of these underground structures are discussed below.

(1) CCW Tunnel

The two CCW tunnels that are to be founded on competent structural backfill connect the CCW Hx structures to the NA. At each end of the tunnels, a gap of 10.16 cm (4 in.) with a water-tight rubber seal is designed between the tunnels and the adjacent structures. The tunnels have a cross-sectional dimension of 2.44 m (8 ft) x 2.44 m (8 ft) and the thickness of the walls, roof and foundation mat is 0.92 m (3 ft). They will be designed and constructed in accordance with ACI 349 with the following material properties:

$f'_c = 27.6 \text{ MPa (4,000 psi)}$ and $f_y = 413.7 \text{ MPa (60,000 psi)}$.

The design-basis loads for the tunnels are dead loads, live loads, hydrostatic fluid pressure loads, soil static pressure loads, dynamic soil pressure due to earthquake, thermal loads, truck loads, and seismic loads. The load combinations for the design are specified in CESSAR-DC Appendix 3.8A. In the analysis, the tunnel was considered as a beam on an elastic foundation and the equivalent static analysis was performed. When the seismic loads were considered in the analysis and design, as discussed in Section 3.7.2 of this report (above), the analysis of the buried tunnels considered the strains (axial and bending) and the associated

stresses due to the effects of seismic wave passage and seismically induced differential movements of the ends of the tunnel. In addition, the groundwater effects were also considered in the design.

During the audit on January 31 through February 1, 1994, the staff raised a concern that in the CESSAR-DC, ABB-CE should provide the tunnel joint details for reinforcing steels. Subsequently, ABB-CE provided the staff with the markups for the locations and details to be provided in future CESSAR-DC amendments. This was part of Confirmatory Item 3.8.4.2-1 of the advance version of the SER. In Amendment V, ABB-CE provided these details, which are acceptable and resolve the applicable part of Confirmatory Item 3.8.4.2-1. On the basis of the discussion and the design calculation audit conducted during January 31 and February 1, 1994, the staff concludes that the procedures for the analysis and design results of the buried tunnels are acceptable.

(2) Buried Cable Tunnels and Conduit Banks

The buried cable tunnels and conduit banks, which are to be founded on competent structural backfill, are generally rectangular in cross-section and will be designed and constructed of reinforced concrete to house the cables and conduit for electrical distribution. According to ABB-CE, the same approach as applies to the design of the CCW pipe tunnel will be used for the design of cable tunnels and conduit banks.

On the basis of the discussion, the staff concludes that the procedures used for the analysis and design of these buried structures are acceptable.

On the basis of its review, the staff concludes that the design of seismic Category I NNI structures (including CCW pipe tunnel) is acceptable and meets the relevant requirements of 10 CFR Part 50, and GDC 1, 2, 4, and 5. This conclusion is based on the following:

- (1) ABB-CE meets the requirements of GDC 1 with respect to assuring that the seismic Category I NNI structures are designed, fabricated, erected, and constructed to quality standards commensurate with its safety function to be performed by meeting the guidelines of RGs and industry standards indicated below.
- (2) ABB-CE meets the requirements of GDC 2 by designing the seismic Category I NNI structures to withstand a severe earthquake that is more severe than the SSEs for all currently license plants (with the exception of the two coastal California sites)

with sufficient margin, and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

- (3) ABB-CE meets the requirements of GDC 4 by designing the NI structure foundation to withstand the effects associated with system and equipment failures, including missiles, pipe whip, and loads associated with a spectrum of LOCAs.
- (4) GDC 5 is not applicable because that the structural systems and components are not shared between units.
- (5) ABB-CE meets the requirements of Appendix B because its QA program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of all the plant seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime conform with established criteria, codes, standards, and specifications acceptable to the regulatory staff. These include meeting the guidelines of RGs 1.69, 1.94, 1.115, 1.142, and 1.143 and industry standards ACI-349-85 and ANSI/AISC N-690-84, "Specifications for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities."

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornados, earthquakes, and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

3.8.5 Foundations

3.8.5.1 NI Structures Foundations

By DSER Open Item 3.8.5-1, the staff requested that ABB-CE submit the design description, assumptions, and criteria for the foundations of all seismic Category I structures in the CESSAR-DC. In CESSAR-DC Amendment R, Section 3.8.5, ABB-CE provided the description, applicable codes, standards, and specifications, loads and loading combinations, design and analysis procedures,

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structural acceptance criteria, material and testing requirements for foundations of seismic Category I structures including the NI structures foundation. As described below, this is acceptable and resolves DSER Open Item 3.8.5-1.

The foundation of the NI structures is a reinforced-concrete mat approximately 3 m (10 ft) in thickness. It has a flat bottom and rests on soil or rock. The top of the NI basemat is located $12.42 \text{ m} \pm 0.3 \text{ m}$ ($40.75 \text{ ft} \pm 1 \text{ ft}$) below the finished grade elevation. CESSAR-DC Section 3.8.5.1 states that the site-specific foundation mat construction procedures will be submitted in accordance with the guidelines of SRP Section 3.8.5. This is included in COL Action Item 3.8-1. ACI 349-85 is used as the design code for the NI structures foundation mat.

The design loads and load combinations are described in CESSAR-DC Appendix 3.8A, Section 5.0. The NI structures foundations are designed to the same loads and load combination as their reinforced concrete superstructures.

The reinforced concrete foundation mat of the NI structures is analyzed and designed for the reactions from the static, seismic, and all other significant loads at the base of the supported superstructures. The foundation mat is modeled as a 3-D finite element structure as an integral part of the NI FEM. The analysis and design of the foundation basemat considers the effects of varying soil properties beneath the foundation, potential mat uplift, construction sequence, and differential settlements. In addition, the basemat is reinforced symmetrically to address the potential cracking from differential settlements. CESSAR-DC Section 3.8.5.4, specifies a settlement monitoring program for the NI foundation mat.

CESSAR-DC Section 3.8.5.5 refers to the structural acceptance criteria of CESSAR-DC Section 3.8.4.5, which is applicable to other seismic Category I structures. By DSER Open Item 3.8.5-3, the staff noted that ABB-CE should provide acceptance criteria regarding the factors of safety against overturning, sliding, and flotation. In CESSAR-DC Appendix 3.8A, Section 5.0, ABB-CE stated that the foundations are also checked against sliding and overturning caused by earthquakes, winds, and tornados and against flotation due to floods in accordance with the guidelines of SRP Section 3.8.5. The loading combinations and the factors of safety against overturning, sliding, and flotation are in agreement with the guidelines of SRP Section 3.8.5. The structural acceptance criteria for the NI foundation mat are in conformance with the guidelines of SRP Section 3.8.5. This is acceptable and resolves DSER Open Item 3.8.5-3.

Section 3.8.5.6 of the CESSAR-DC states that epoxy-coated reinforcing steel is used wherever a corrosive environment is encountered. Other criteria for materials, quality control, and construction techniques are described in CESSAR-DC Section 3.8.4. The staff evaluation appears in Section 3.8.4 of this report.

Before issuing the DSER, ABB-CE stated in CESSAR-DC Section 3.8.5.4, that the analysis and design of the foundations require site-specific soil properties. ABB-CE also stated that the detailed design of the foundations and the verification of their acceptability for sliding and overturning shall be performed on a site-specific basis. In the DSER, the staff stated that this response is unacceptable because it is not consistent with the approach taken in CESSAR-DC Sections 2.5.2.5.2, 3.7, and 3.8.1 through 3.8.4, in which generic site conditions are assumed and the structures are designed accordingly before a specific site is selected. The staff also stated that ABB-CE must provide a clear description of the design, assumptions, and criteria used for the foundation mats for all Category I structures including the containment and ISs. The staff also required that design analysis of the foundation mats for the NA and containment and ISs be completed and referenced in the CESSAR-DC. Additionally, the staff noted that construction loadings for the heavy foundation mats may cause unacceptable construction-related stress conditions, depending on site-specific soil conditions. This was DSER Open Item 3.8.5-2.

In the October 28 and 29, 1993, audit at Duke Engineering Services Co., the staff audited the structural design and detailing calculations for the NI structures foundation mat. The main purpose of the audit was to review and discuss the design calculations, evaluate the design adequacy, and obtain some insights on the design process in order to incorporate these into the CESSARDC. The design review of the NI structures basemat gave the staff confidence that the overall design is acceptable and that the critical dimensions and reinforcing bar patterns, as described in Appendix 3.8B to CESSAR-DC, are adequate.

ABB-CE has completed the design and stability calculations for the NI foundation mat. ABB-CE has also completed design analysis, including evaluation of sliding and overturning stability, of the foundation mat for the NI structures. This design information is referenced in Appendix 3.8B to the CESSAR-DC and is made available in an auditable form. This is acceptable and resolves DSER Open Item 3.8.5-2.

By DSER COL Action Item 3.8.5-1, the staff noted that COL applicants should submit the site-specific foundation mat construction procedures to the staff for review and approval. CESSAR-DC Section 3.8.5.1 incorporates the

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above information. This is acceptable. This is included in COL Action Item 3.8-1.

On the basis of its review, the staff determines that sufficient information is provided in the CESSAR-DC in accordance with the guidelines of SRP Section 3.8.5. The staff concludes that the design of the NI structures foundation is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDC 1, 2, 4, and 5. The conclusion is based on the following:

- (1) ABB-CE meets the recommendations of SRP Section 3.8.5 and the requirements of 10 CFR 50.55a and GDC 1 by ensuring that the NI structural foundations are designed, fabricated, erected, constructed, tested, and inspected to the quality standards commensurate with its safety function to be performed by meeting the guidelines of RGs and industry standards.
- (2) ABB-CE meets the requirements of GDC 2 by designing the NI structures foundation to withstand a severe earthquake that is more severe than the SSEs for all currently licensed plants (with the exception of the two coastal California sites) with sufficient margin, and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.
- (3) ABB-CE meets the requirements of GDC 4 by designing the NI structures foundation to withstand the effects associated with system and equipment failures, including missiles, pipe whip, and loads associated with a spectrum of LOCAs.
- (4) GDC 5 is not applicable because the structural systems and components are not shared between units.
- (5) ABB-CE meets the requirements of Appendix B because their QA program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of the NI structures foundation to account for anticipated loadings and postulated conditions that may be imposed on the structures during their service lifetime conform with established criteria, codes, standards, and RGs acceptable to the staff. These include meeting the positions of RGs 1.69, 1.94, 1.115 and 1.142 and industry standards ACI 349-85 and ANSI/AISC N690-84.

The use of these criteria as defined by the applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials and quality control programs; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the structures, the foundations of the NI structures will withstand the specified conditions without impairment of structural integrity or of the performance of required safety functions.

3.8.5.2 Category I NNI Foundations

In the System 80+ design, ABB-CE employs separate reinforced-concrete mat foundations for seismic Category I NNI structures such as the DFSS and CCW Hx structures. The plan dimensions of the foundation mats for the DFSS and CCW Hx structure, as described in CESSAR-DC Section 3.8.5, are 19.2 m x 13.4 m (63 ft x 44 ft) and 33.5 m x 13.4 m (110 ft x 44 ft), and the minimum thickness of these two foundation mats are 0.7 m (2.25 ft) and 1.2 m (4 ft), respectively.

As described in CESSAR-DC Section 3.8.5, Amendment U, the reinforced concrete foundation mats of the DFSS and CCW Hx structures were analyzed and designed for the reactions caused by static, seismic and all other design-basis loads at the base of the superstructures supported by the foundations. The foundation mat together with the superstructures of the DFSS was modeled as a 3-D FEM and analyzed by computer code ANSYS. The foundation mat of the CCW Hx structure was analyzed by hand calculation. Both analyses considered the envelopes of the seismic loads calculated for all 12 site conditions and three control motions discussed in Section 3.7.2.2 of this report; and the results obtained from the analyses together with the other design loads, were used for the foundation design. The analyses and design also considered the effects of varying soil properties beneath a specific foundation mat and the effects of construction sequence, with particular emphasis on differential settlement of the foundation. To monitor the settlements of the foundation after the completion of construction, settlement monitoring devices will be installed. For the foundation design, the ACI-349-85 code was used. The acceptance of the ACI-349 code is discussed in Section 3.8.4.1 of this report.

As described in CESSAR-DC Sections 3.8.4.4 and 3.8.5 and Appendix 3.8A, Amendment U, major materials used in the design and construction of the DFSS are concrete, reinforcing bars, and structural steel. Cement for concrete will be of Type I or II conforming to "Standard Specifications for Portland Cement," ASTM-C150. Aggregates for

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concrete will conform to "Standard Specification for Concrete Aggregate," ASTM C33. Water used in mixing concrete will be clean and free from injurious amounts of oil, acids, alkalis, salts, organic materials or other substances that may be deleterious to concrete or steel. The proposed mixing water properties will be compared with distilled water by performing the tests described in CESSAR-DC Section 3.8.4.6.1. Admixtures, if used, will conform to the applicable ASTM standard in CESSAR-DC Section 3.8.4.6.1. In order to prevent corrosion of reinforcing bars, the combined chloride content of the admixtures and mixing water will not exceed 250 ppm. The ingredient materials will be stored in accordance with the recommendations of ACI-304 and the concrete mixes will be designed in accordance with ACI-301. Reinforcing steel will consist of deformed reinforcing bars conforming to ASTM A615, Grade 60 or ASTM A706, Grade 60. The fabrication and fabrication tolerances of reinforcing bars will be in accordance with CRSI MSP-1, "Manual of Standard Practice." The placement of reinforcing bars, including spacing of bars, concrete protection of reinforcement, splicing of bars, and field tolerances will be in accordance with ACI-349-85. Epoxy-coated reinforcing steel bars are used wherever a corrosive environment is encountered. For calculating the development length, CESSAR-DC Appendix 3.8A, Section 6.2.1.1.1, states that the required splice length given in ACI-349 Section 12.2.2 shall be increased using the factors provided in ACI-318 Section 12.2.4.3.

The quality control of materials will be in accordance with the relevant ASTM Specifications and the overall QA program described in CESSAR-DC Chapter 17, as supplemented by the special provisions of ACI-349-85. The strength of the construction materials for the foundations are as follows:

$$f'_c = 27.6 \text{ MPa (4,000 psi)} \quad \text{for concrete}$$

$$f_y = 413.7 \text{ MPa (60,000 psi)} \quad \text{for reinforcing steel}$$

From this discussion and the design calculation audit done on January 31 and February 1, 1994, the staff concludes that the foundation design of these two buildings are acceptable. In addition to satisfying the requirements for the design loads and combined load conditions, an evaluation was performed to check the dynamic stability (sliding, overturning and floatation) of the foundations against the seismic loads. During the design calculation audit done on January 31 and February 1, 1994, the staff found that the safety coefficients against dynamic stability for the DFSS and CCW Hx structures are higher than 1.1 as specified in SRP Section 3.8.5. This is acceptable.

On the basis of its review, the staff concludes that the design of the DFSS and CCW Hx structure foundations are acceptable and meets the relevant requirements of 10 CFR Part 50 and GDC 1, 2, 4, and 5. This conclusion is based on the following:

- (1) ABB-CE meets the requirements of GDC 1 with respect to assuring that the seismic Category I foundations are designed, fabricated, erected, constructed, tested, and inspected to the quality standards commensurate with its safety function to be performed by meeting the guidelines of RGs and industry standards.
- (2) ABB-CE meets the requirements of GDC 2 by designing the seismic Category I foundations to withstand a severe earthquake that is more severe than the SSEs for all currently licensed plants (with the exception of the two coastal California sites) with sufficient margin, and the combinations of the effects of proper combined load conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.
- (3) ABB-CE meets the requirements of GDC 4 by designing the NI structures foundation to withstand the effects associated with system and equipment failures, including missiles, pipe whip, and loads associated with a spectrum of LOCAs.
- (4) GDC 5 is not applicable because the structural systems and components are not shared between units.
- (5) ABB-CE meets the requirements of Appendix B because its QA program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis and design, and construction of all the plant seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime conform with the established criteria, codes, standards, and specifications acceptable to the staff. These include meeting the guidelines of RG 1.142 and industry standard ACI-349-85.

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornados, earth-

quakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or of the performance of required safety functions.

3.9 Mechanical Systems and Components

The staff reviewed CESSAR-DC Section 3.9 in accordance with SRP Sections 3.9.1 through 3.9.6. The review pertained to the structural integrity and functional capability of various safety-related mechanical components in the System 80+ plant. The review was not limited to ASME Code components and supports, but extended to such other components as control rod drive mechanisms (CRDMs), certain reactor internals, and any safety-related piping designed to industry standards other than the ASME Code. The staff reviewed such issues as load combinations, allowable stresses, methods of analysis, summary of results, preoperational testing, and IST of pumps and valves. To find these acceptable, the staff must conclude that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The staff reviewed the information in CESSAR-DC Section 3.9.1 relative to the design transients and methods of analysis used for all seismic Category I components, component supports, core support structures, and reactor internals designated as Class 1, 2, 3, and CS under ASME Code, Section III, and those components not covered by the code. It reviewed the assumptions and procedures used for the inclusion of transients in the design and fatigue evaluation of ASME Code Class 1 and CS components. It also reviewed the computer programs used in the design and analysis of seismic Category I components and their supports, as well as experimental and inelastic analytical techniques.

Design Transients

The staff reviewed the design transients specified in CESSAR-DC Section 3.9.1 and elsewhere that are to be used in the design and fatigue analysis of ASME Code components for the 60-year plant design life. The staff found that (1) in CESSAR-DC Sections 3.9.1.1 and 3.9.3.1.1, ABB-CE specifies that the design transients for ASME Code, Section III, Class 1 components and supports are provided in CESSAR-DC Table 3.9-1, (2) in CESSAR-DC Section 3.9.3.1.3, ABB-CE specifies that the system operating conditions for ASME Code, Section III,

Class 2 and 3 components and supports are based on the design transients of Table 3.9-1 together with other specific auxiliary system conditions, and (3) in CESSAR-DC Section 3.9.5.2, ABB-CE specifies the loading conditions that are considered in the design of ASME Code Class CS core support and reactor ISs.

By RAI Q210.46, the staff raised a concern relating to the adequacy of margins available in the ASME Code Class 1 fatigue design curves to ensure a 60-year design life. On the basis of the limited data available, the staff noted that these margins may not be sufficient to account for variations in the original fatigue test data because of various environmental effects that may be exacerbated during the 60-year design life. The staff asked ABB-CE to consider such effects in the design of ASME Code Class 1 systems, components, and equipment. ABB-CE committed to comply with revisions to the existing fatigue curves or modifications to current fatigue design and evaluation methodologies that may be incorporated in future editions of the code in response to the same concern. In the DSER, the staff noted that ABB-CE shall modify its commitment to state that it will perform fatigue analyses in accordance with NRC-approved methods to address environmental concerns. This was DSER Open Item 3.9.1-1. In response, in CESSAR-DC Section 3.9.1.1 (Amendment R), ABB-CE stated that observations of significant environmental degradation in LWR environments are primarily related to high strain ranges, slow strain rates, high oxygen content of primary water, high sulfur content of carbon and low alloy steels, and low flow rate conditions. The absence of any one of these conditions is sufficient to preclude any significant environmental degradation of the fatigue behavior of materials exposed to typical PWR primary coolant environment. ABB-CE further noted that, the System 80+ plant is not exposed to high oxygen content environments at elevated temperatures, and no carbon or low alloy steel is directly exposed to the primary coolant. On this basis, ABB-CE concluded that no significant environmental degradation of System 80+ components will occur. This is acceptable and resolves DSER Open Item 3.9.1-1.

By RAI Q210.46, the staff also observed that the transients and the number of cycles for the transients listed in CESSAR-DC Table 3.9-1 specified for the 60-year design life of the System 80+ plant are the same as those listed for the 40-year design life of the System 80 plant. In CESSAR-DC Section 3.9.1.1 of the early amendments, ABB-CE stated that the number of cycles was based on a frequency-of-occurrence type of evaluation. ABB-CE stated that although both plants use the same list of transients and have the same number of cycles for each of the transients, they have design lives differing by 20 years. By DSER Open Item 3.9.1-2, ABB-CE was asked to justi-

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fy the use of the same number of cycles for the System 80+ design as was used for the System 80 design. In Amendment N to the CESSAR-DC, ABB-CE revised Table 3.9-1 to account for the number of cycles applicable to 60-year plant life. The staff finds the revised Table 3.9-1 acceptable. On this basis, DSER Open Item 3.9.1-2 is resolved.

By DSER Open Item 3.9.1-3, the staff requested that ABB-CE justify limiting its consideration of design transients to those not requiring forced shutdown in the design of ASME Code Class CS components and reactor ISs as described in CESSAR-DC Section 3.9.5.2, Item G. In response, ABB-CE revised CESSAR-DC Section 3.9.5.2 (Amendment N) to include all anticipated transient loadings in Table 3.9-1, and appropriate design-basis pipe break (DBPB), secondary side break, and LOCA loads. This revision is acceptable and resolves DSER Open Item 3.9.1-3.

ABB-CE meets GDC 14 and 15 by demonstrating that the design transients and resulting loads and load combinations with appropriate specific design and service limits that ABB-CE used for the design of ASME Code, Class 1 and CS components and supports and reactor internals provide a complete basis for the design of the RCPB for all conditions and events as specified in CESSAR-DC Table 3.9-1 over the service lifetime of the plant.

Computer Programs

The staff reviewed the information provided in CESSAR-DC Sections 3.9.1.2.1 through 3.9.1.2.3 relating to computer programs used in the stress analysis of (1) RCS, (2) Code Class CS internals and fuel and control element drive mechanism (CEDM), and (3) non-NSSS structures and components, respectively.

By RAI Q210.47, the staff noted that SRP Section 3.9.1, Revision 2, Item II.2, defines the information to be provided to demonstrate the applicability and validity of computer programs to be used in dynamic and static analyses to demonstrate the structural and functional integrity of seismic Category I, and ASME Code and non-code items. The information provided in an early version of CESSAR-DC Section 3.9.1.2 was not totally in accordance with the guidelines of SRP Section 3.9.1, Revision 2. The staff asked ABB-CE to provide the computer program information required by SRP Section 3.9.1, Revision 2. The staff noted that a program in the public domain is insufficient for satisfying the guidelines in SRP Section 3.9.1, Revision 2.

ABB-CE responded that, as stated in CESSAR-DC Section 3.9.1.2, the versions of public domain computer codes

used by ABB-CE have been extensively verified to supplement any existing public documentation. Each code (e.g., MDC STRUDL, CE MARC) has had its quality assured in accordance with the QA procedures in force at the time of the documented verification. Documentation for all of the codes described in CESSAR-DC Section 3.9.1.2 is available for NRC audit upon request. The QA calculations for each code contain (1) the author's name, (2) a computer code certificate containing necessary information such as the version, installation date, and facility on which the code executes, and (3) the solutions to appropriate test problems. As discussed in Section 3.12.4 of this report, the staff performed an independent confirmatory piping stress analysis of representative piping systems in the System 80+ standard plant. The results of the confirmatory analysis demonstrate ABB-CE's computer programs to be adequate with acceptable accuracy.

In the DSER, the staff noted that ABB-CE should add a description of the SASSI program in a future revision of the CESSAR-DC. This was DSER Confirmatory Item 3.9.1-1. In Amendment O to CESSAR-DC, ABB-CE described SASSI (Appendix 3.7B). On this basis, DSER Confirmatory Item 3.9.1-1 is resolved.

By DSER Open Item 3.9.1-4, the staff noted that no computer programs were identified in CESSAR-DC Section 3.9.1.2.3 for the stress analysis of non-NSSS mechanical components, and requested that ABB-CE revise CESSAR-DC Section 3.9.1.2.3 to identify computer programs used in stress analyses for non-NSSS mechanical components. In Amendment N to CESSAR-DC, ABB-CE deleted Section 3.9.1.2.3 and stated that the computer programs identified in Section 3.9.1.2.1 are applicable to RCS as well as to non-NSSS mechanical components. CESSAR-DC Section 3.9.1.2.1 also states that if computer codes not listed in CESSAR-DC are used, they will be compared to NRC-benchmarked or approved codes. On this basis, DSER Open Item 3.9.1-4 is resolved.

Experimental Stress Analysis

By DSER Open Item 3.9.1-5, ABB-CE was asked to clarify CESSAR-DC Section 3.9.1.3 (1) to state if experimental stress analyses are used anywhere in the entire System 80+ plant design and (2) if they are used, to commit to performing these analyses in accordance with the SRP Section 3.9.1, Item II.3, criteria. In Amendment N to CESSAR-DC Section 3.9.1.3, ABB-CE stated that when experimental stress analysis is used, it is performed in accordance with Appendix II of ASME Code, Section III, Division I. This is in compliance with the guidelines of SRP Section 3.9-1, III.3, and therefore, resolves DSER Open Item 3.9.1-5.

Level D Service Limits

The staff reviewed the information provided in CESSAR-DC Sections 3.9.1.4.1 and 3.9.1.4.2 relating to Level D service limits for seismic Category I RCS (i.e., NSSS) and non-NSSS items, respectively.

The loads identified in an early version of CESSAR-DC Section 3.9.1.4.1 for seismic Category I NSSS items did not include forces associated with postulated pipe breaks that were specified in CESSAR-DC Section 3.6.2.2.1 for piping systems subject to LBB evaluation. These systems included reactor coolant loop, SL, shut down coolant line, SI line, and MSL inside the containment. By DSER Open Item 3.9.1-6, the staff noted that ABB-CE should consider loads due to DBPB as well as branch line breaks not eliminated by LBB. However, based on NRC approval of the application of the LBB methodology to these piping systems (as discussed in Section 3.6.3 of this report), forces associated with pipe breaks in these systems need not be included in the Level D analyses. Among these forces are pipe thrust forces at the break locations, resultant subcompartment differential pressurization forces, and asymmetric pressurization forces acting on the reactor vessel and internals. In CESSAR-DC Table 3.9-2 for loading combinations, ABB-CE considers in its analysis only those pipe rupture loads resulting from line breaks not eliminated by LBB. This is acceptable. On this basis, DSER Open Item 3.9.1-6 is resolved.

SRP Section 3.9.1, Revision 2, Item III.4, provides guidelines for methods of analysis, including elastic-plastic methods, used in the evaluation of Level D service conditions. By RAI Q210.48, the staff asked ABB-CE to state where inelastic methods of analyses were used in seismic Category I RCS items as implied in CESSAR-DC Section 3.9.1.4.1. ABB-CE responded to RAI Q210.48 that inelastic methods were not used in analyzing these items.

CESSAR-DC Section 3.9.1.4.1 states that for the evaluation of RCS faulted condition, the pipe break load analysis procedure considers only those breaks not eliminated by LBB. Based on the evaluation in Section 3.6.3 of this report, LBB is applicable for the System 80+ design and certain design-basis LOCA loads need not be included in the faulted condition loading combination for the design of piping and component supports. On this basis, DSER Open Item 3.9.1-7 resolved.

Originally, in CESSAR-DC Section 3.9.1.4.2, ABB-CE stated that in exceptions to linear elastic models for evaluating faulted conditions, the maximum allowable strain limits from accepted standards will be satisfied. By RAI Q210.50, the staff asked ABB-CE to list and justify

the standards and the strain limits to be used. ABB-CE responded to RAI Q210.50 by committing to revising CESSAR-DC Section 3.9.1.4.2 to specify that pipe-rupture restraint and energy-absorbing members are the exceptions to the use of linear elastic models. In CESSAR-DC Sections 3.6.2.3.2.4 and 3.6.2.3.2.5, respectively, as modified by ABB-CE's responses to RAIs Q210.31 and Q210.32, ABB-CE gives the allowable stresses and design criteria for these members.

The staff concludes that ABB-CE has demonstrated compliance with the requirements of GDC 1, 2, 14, and 15 and guidelines of SRP Section 3.9.1, as discussed below.

ABB-CE meets GDC 14 and 15 by demonstrating that the design transients and resulting loads and load combinations with the appropriate specific design and service limits for designing ASME Code, Class 1 and CS components and supports and reactor internals are a complete basis for the design of the RCPB for all anticipated conditions and extremely low-probability events expected over the service lifetime of the plant.

ABB-CE meets GDC 2 and Appendix A to 10 CFR Part 100, by including seismic events in design transients that serve as the design basis for withstanding the effects of natural phenomena.

ABB-CE meets Appendix B to 10 CFR Part 50, and GDC 1, by submitting information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I structures designated as ASME Code, Class 1, 2, 3, and CS and those not covered by the Code within the present state-of-the-art limits and by having design control measures that are consistent with the applicable guidelines of SRP Section 3.9.1. This is acceptable for ensuring the quality of the computer programs. If the COL applicant opts to use computer programs different from those used by ABB-CE for the design of any safety-related item with the exception of piping systems, the guidelines of SRP Section 3.9.1 must be met for such programs.

3.9.2 Dynamic System Analysis and Testing

The staff reviewed the methodology, testing procedures, and dynamic analyses used by ABB-CE to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loadings. The staff reviewed (1) the piping vibration, thermal expansion, and dynamic effects testing, (2) the seismic subsystem analysis methods, (3) the dynamic responses of structural components in the reactor caused by steady-state and operational flow transient conditions, (4)

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the flow-induced vibration testing of reactor internals to be conducted during the preoperational and start-up test program, and (5) the dynamic analysis methods used to confirm the structural design adequacy and functional capability of the reactor internals and piping attached to the reactor vessel when subject to loads from applicable LOCAs and DBPBs in combination with an SSE.

3.9.2.1 Piping Vibrations, Thermal Expansion, and Dynamic Effects

In the early amendments to CESSAR-DC Section 3.9.2.1, ABB-CE described the piping vibration, thermal expansion, and dynamic effects testing to be conducted during a preoperational testing program. These tests ensure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent. By RAI Q210.51, the staff noted that the test program described in CESSAR-DC Section 3.9.2.1 is not fully consistent with the guidelines in SRP Section 3.9.2, Revision 1, Item II.1. To meet these guidelines, the staff asked ABB-CE to commit to conduct the program in accordance with ANSI/ASME OM-3 and OM-7 and to identify the piping systems to be included in the test program. ABB-CE responded to RAI Q210.51 by committing to conduct the program in accordance with ANSI/ASME OM-3 and OM-7 and identified typical piping systems to be tested.

The staff noted in the DSER, however, that although the stresses on which limits on piping displacements during steady-state vibration were to be based (as described in CESSAR-DC Section 3.9.2.1.1) were the same as those previously found to be acceptable to the staff for 40-year design-life plants, ABB-CE did not justify the applicability of these stresses to a 60-year design-life plant.

Accordingly, by DSER Open Item 3.9.2.1-1, ABB-CE was asked to identify the specific version of ASME OM-3 and OM-7 standards, and to justify the applicability of their stresses on which limits on piping displacements during steady-state vibration are to be based for the 60-year design life of the System 80+ plant. Subsequently, CESSAR-DC Section 3.9.2.1 states that the preoperational test program will be conducted in accordance with ASME OM-3 and OM-7 standards which are specified in CESSAR-DC Table 1.8-7 to be the 1990 version through 1992 addenda. Furthermore, ABB-CE stated that the use of the preceding OM-3 standard would limit the piping vibration displacements to 80 percent of the minimum alternating stresses in the ASME Code fatigue curves and material dependent stress reduction factors. These factors account for the differences between the number of cycles for which data are provided in the ASME Code fatigue curves. The 80 percent reduction is consistent with the

OM-3 standard and the material stress reduction factors are acceptable to account for the number of the cycles consistent with a 60-year design life. This is acceptable and resolves DSER Open Item 3.9.2.1-1.

The staff noted, in DSER Open Item 3.9.2.1-2, that in CESSAR-DC Section 3.9.2.1, ABB-CE states that the testing program was applicable to ASME Code, Section III, Class 1, 2, and 3 piping systems. ABB-CE did not, however, identify testing applicable to non-ASME or other systems. In Amendment N to CESSAR-DC, ABB-CE revised Section 3.9.2.1 to include non-ASME and other systems in the test programs in accordance with the guidance of SRP Section 3.9.2, Item I.1. On this basis, DSER Open Item 3.9.2.1-2 is resolved.

On the basis of its review of CESSAR-DC Section 3.9.2.1, as set forth above, the staff concludes that ABB-CE meets GDC 14 and 15 with respect to the design and testing of the RCPB. This gives reasonable assurance that rapidly propagating failure and gross rupture will not occur as a result of vibratory loadings. In addition, the testing ensures that design conditions will not be exceeded during normal operation, including anticipated operational occurrences, by having an acceptable vibration, thermal expansion, and dynamic effects test program that will be conducted during startup and initial operation of specified high- and moderate-energy piping, including all associated restraints and supports. The tests give adequate assurance that the piping and piping supports will be designed to withstand vibrational dynamic effects as a result of valve closures, pump trips, and other operating modes associated with the design-basis flow conditions. In addition, the tests ensure that adequate clearances and free movement of snubbers will exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. For the planned tests, loads similar to those experienced during transient and normal reactor operations will be developed.

3.9.2.2 Seismic Analysis of Safety-Related Mechanical Equipment

The staff reviewed areas relating to seismic subsystem analysis on the basis of the information submitted in CESSAR-DC Sections 3.7.3 and 3.9.2.2. The review was conducted in accordance with the guidelines of SRP Section 3.7.3 which are essentially the same as those of SRP Section 3.9.2. In DSER Section 3.7.3, the staff reported its review of the seismic subsystem analysis for both structural and mechanical areas. Although the same seismic analysis methodology may be used for both structural and mechanical subsystems, the specific considerations and acceptance criteria may be different. For this reason, the staff decided to report its review in this section

of issues related to mechanical subsystems, equipment and their supports. The staff continues to report its review of structural subsystems, including cable tray, conduit and HVAC support systems, intake structure, tanks, and tunnels in Section 3.7.3 of this report.

Seismic Analysis Methods

In the early amendments to CESSAR-DC Section 3.7.3, ABB-CE committed to analyzing seismic Category I structures, subsystems, and components by the response spectrum or time-history method as described in CESSAR-DC Section 3.7.2.1.1 or by an equivalent static method as described in CESSAR-DC Section 3.7.3.5. These analysis methods are in accordance with the criteria in SRP Section 3.7.2, Item II.1, referenced by SRP Section 3.7.3, Item II.1. However, neither CESSAR-DC Section 3.7.3.1 nor 3.7.2.1.1 contains an explicit commitment to consider the items identified in SRP Section 3.7.2, Revision 2, Items II.1.(a)(iii) and (iv).

By RAI Q210.38, the staff asked ABB-CE to commit to these SRP Section 3.7.2, Revision 2, criteria. In response to RAI Q210.38, ABB-CE did not commit to these criteria but stated that these criteria were used in System 80+ modeling procedures. This is reflected in the criteria of CESSAR-DC Section 3.7.3.3 that the dynamic analysis models be generated to accurately evaluate the dynamic behavior of the component. This response was not totally acceptable since no acceptance criteria were specified for determining if the models in the dynamic analyses were acceptable to accurately evaluate the component behavior.

By RAI Q210.38 the staff also asked ABB-CE to submit seismic subsystem (modeling) criteria and to demonstrate that the number of mass points and DOFs in their models were in accordance with the criteria of SRP Section 3.7.2, Revision 2, II.1.a(iii). The staff was concerned about the adequacy of the RCS seismic subsystem analyses model in CESSAR-DC Figure 3.7-32. ABB-CE responded to this request by referencing the technical approach described in the response to RAI Q210.26. In the response, ABB-CE stated that a System 80+ DSDG was to be prepared to ensure that the final design would be consistent with the design basis and methodology in CESSAR-DC.

In DSER Open Item 3.7.3-1, the staff noted that information related to piping analytical methodologies was to be included in the DSDG - which was eventually to become a part of the CESSAR-DC - and that the staff evaluation would not be completed until after the staff completes its review of the DSDG. Subsequently, the staff completed its review of the DSDG, and portions noted by the staff as necessary for its safety evaluations were extracted, clari-

fied or modified as necessary, and included in CESSAR-DC Appendix 3.9A.

Relative to the requests concerning the model in CESSAR-DC Figure 3.7-32, based on information provided by ABB-CE in response to RAI Q210.33 (SEE Section 3.7.2 of this report), it was determined that the fundamental frequencies of the components of the RCS was sufficiently high (>100 Hz) that the RCS would respond essentially as coupled rigid bodies in response to seismic loading. Accordingly, the number of degrees of freedom and number of modes of the RCS system associated with the model in CESSAR-DC Figure 3.7-32 are sufficient to accurately predict the dynamic behavior of the RCS subsystem under seismic loadings. On this basis, the staff finds that concerns regarding the model in CESSAR-DC Figure 3.7-32 are resolved.

Relative to the generic concern regarding the lack of acceptance criteria for the acceptability of seismic subsystem models to accurately predict component responses, ABB-CE included, in Appendix 3.9A, Section 1.4.3.2.1.4 (Amendment Q), analysis criteria for modal frequency cutoff and rigid range acceleration effects. These criteria meet the guideline of the 10-percent increase in responses of SRP Section 3.7.2, II.1.a(iii) and (iv). On this basis, DSER Open Item 3.7.3-1 is resolved.

Determination of Number of Earthquake Cycles

In CESSAR-DC Section 3.7.3.2 (Amendment E), ABB-CE stated that, for System 80+ plants, seismic Category I subsystems, components, and equipment are designed to a total of 200 full-load earthquake cycles for a 60-year design life. This was the same number of cycles specified for the CE System 80 plant which has a 40-year design life, but ABB-CE did not justify that the 200 earthquake cycles specified for the System 80+ design should be the same as for the System 80 design. Also, the staff's review of Section 7.0 of the draft DSDG revealed the discrepancy that ABB-CE specified 200 cycles for NSSS piping but 75 cycles for Class 1 piping other than NSSS piping.

By DSER Open Item 3.7.3-2, ABB-CE was asked to (1) justify the use of the same number of earthquake cycles for both the 60-year System 80+ design life and the 40-year System 80 design life and (2) explain the discrepancy between the information in CESSAR-DC 3.7.3.2 and Section 7.1.4.3.2.1.8 of the draft DSDG.

By Amendment R to CESSAR-DC Section 3.7.3.2, ABB-CE revised the number of earthquake cycles for the System 80+ design to be a total of two SSE events with 10 maximum stress cycles per event. In SECY-93-087, the staff recommended the elimination of the OBE in the

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design process on the basis that it would not result in a significant decrease in the overall plant safety margin. The detailed basis for the staff's recommendation is discussed in Section 3.1.1 of this report, which also defines acceptable number of earthquake cycles or alternatives in fatigue evaluations. The revised CESSAR-DC Section 3.7.3.2 (Amendment R) is consistent with the staff position discussed in Section 3.1.1 of this report. This is, therefore, acceptable.

In addition, contrary to earlier expectations, the entire DSDG has not been included in CESSAR-DC. Instead, portions identified by the staff as necessary for its safety evaluations were extracted, clarified, and modified as necessary and included in CESSAR-DC Appendix 3.9A. Previous draft DSDG Section 7.1.4.3.2.1.8 is contained in current CESSAR-DC Appendix 3.9A, Section 1.4.3.2.1.7 and is modified to reference CESSAR-DC Section 3.7.3.2 for the cyclic load basis for fatigue evaluations. This is acceptable and resolves DSER Open Item 3.7.3-2.

Procedures Used for Analytical Modeling

In the early amendments to CESSAR-DC Section 3.7.3.3, ABB-CE submitted information on modeling techniques in general and detailed information on pipe modeling. For modeling in general, ABB-CE stated that each model will be sufficiently detailed to accurately evaluate the dynamic behavior of the component. For piping, ABB-CE specified criteria for spacing piping mass point and criteria for representing valve and other concentrated weight. These criteria include provisions for the effects due to offset of the mass from a pipe center line.

By RAI Q210.38, the staff asked ABB-CE to clarify the seismic subsystem modeling techniques in CESSAR-DC Section 3.7.3.3. ABB-CE committed to use the criteria of SRP Section 3.7.2, Revision 2, Item II.3.b and SRP Section 3.7.2, Revision 2, Item II.1.a.(iii) for dynamic decoupling of subsystems and for modeling criteria, respectively. The use of these criteria for the purposes stated are in accordance with SRP Section 3.7.3, Items II.1 and II.3 requirements, and hence, are acceptable.

In the early amendments to CESSAR-DC Section 3.7.2.1.2.2, ABB-CE described the seismic subsystem analysis performed for the SL. The analysis involves decoupling the SL from the major RCS components to which it is attached. Although decoupling the SL from the major RCS components may be acceptable, ABB-CE did not justify doing this. The staff requested that ABB-CE submit information, in RAI Q210.34, regarding modeling of the SL. ABB-CE committed to comply with modeling acceptance criteria of SRP Section 3.7.2, Revision 2, Item II.1.a.(iii).

By Amendment Q, ABB-CE submitted details of a typical model for the seismic analysis of the SL in CESSAR-DC Figure 3.7-25. The model contains 31 mass points each having 3 translational DOFs. The staff finds that this model satisfies the criteria of SRP Section 3.7.2, Revision 2, Item II.1.a.(iii) and is acceptable. On this basis, DSER Confirmatory Item 3.7.3-1 is resolved.

Basis for Selection of Frequencies

In the early amendments to CESSAR-DC Section 3.7.3.4, ABB-CE provided information on the design approach to avoid resonance of components and subsystems. In this approach, within practical limitations, the fundamental frequencies of the components and subsystems are designed to be "sufficiently removed" from the resonant range. If resonance does occur, the resonance should be accounted for in the analysis.

By RAI Q210.39 the staff asked ABB-CE to demonstrate that the design approach in CESSAR-DC Section 3.7.3.4 was in accordance with SRP Section 3.7.3, Revision 2, Item II.4, which describes quantitative guidelines for fundamental frequencies of components and subsystems in contrast to the qualitative "sufficiently removed" criterion in ABB-CE's approach. More importantly, regardless of the design approach of avoiding resonance, the seismic response of components and subsystems, and not just resonant response, should be included in the analysis of the components and subsystems. Accordingly, by DSER Open Item 3.7.3-3, the staff noted that ABB-CE should revise CESSAR-DC Section 3.7.3.4 to account for the seismic response of SSCs in their design and not just the resonant response.

Subsequently, in CESSAR-DC Section 3.7.3.4, ABB-CE incorporated the modified design approach for resonance by stating that the seismic design will account for the seismic response of structures, subsystems, and components. On this basis, DSER Open item 3.7.3-3 is resolved.

Analysis Procedure for Damping

CESSAR-DC Section 3.7.3.15 provides procedures for damping by referring to other sections in the CESSAR-DC, such as Sections 3.7.2.15 and 3.7.3.14. Additional information on damping is given provided in CESSAR-DC Section 3.7.3.6 and Table 3.7-1.

Considering the information noted in the preceding paragraph and ABB-CE's response to RAIs Q210.36 and Q210.37, ABB-CE's procedures for assessing damping for structures, subsystems, and components are in accordance with (1) CESSAR-DC Table 3.7-1, modified as committed

to in the response to RAI Q210.37, and (2) CESSAR-DC Section 3.7.2.15 when composite modal damping is used. In response to RAI Q210.37, ABB-CE committed to revising CESSAR-DC Table 3.7-1 to specify that the use of ASME Code Case N-411-1 damping values are subject to all the conditions identified in RG 1.84. These procedures are in compliance with RGs 1.61 and 1.84 guidelines and SRP Section 3.7.3, Item II.5, criteria and, hence, acceptable. Verification of the revision of CESSAR-DC Table 3.7.1 as committed to in the ABB-CE response to RAI Q210.37 was identified as DSER Confirmatory Item 3.7.3-2.

By Amendment N, ABB-CE revised CESSAR-DC Table 3.7.1 to specify that when the response spectrum method of analysis is used, damping values may be based on Code Case N-411-1 as limited by RG 1.84. This resolves DSER Confirmatory Item 3.7.3-2.

Three Components of Earthquake Motion

CESSAR-DC Section 3.7.3.3 (which references CESSAR-DC Section 3.7.2.6) states that the total response of substructures for the response spectrum method of analysis for the three independent components of earthquake motion (two horizontal and one vertical) is obtained by combining the separate responses by the SRSS method. For the time-history methods, these statistically independent earthquake motions are used as input and the total response is obtained by algebraically summing the responses in the time domain. This is in accordance with the criteria of SRP Section 3.7.3, Item II.6, and, therefore, is acceptable.

Combination of Modal Responses

CESSAR-DC Section 3.7.3.7 states that for the response spectrum method of analysis (1) the SRSS method of combination is used for separated modes, and (2) the RG 1.92 method of combination is used for closely spaced modes. These provisions are in accordance with the criteria of SRP Section 3.7.3, Item II.7 and, hence, are acceptable.

Analytical Procedures for Piping

In RAI Q210.40, the staff requested that ABB-CE explain how any significant seismic response of supports and equipment will be included in the seismic analysis of piping, and specifically address how piping input loadings are developed for flexible equipment and supports. ABB-CE responded that equipment and piping supports are assumed to be significantly more rigid than the piping itself. Upon completion of the piping analysis and subsequent support design and equipment procurement, support

and equipment stiffness will be included in the piping analysis if a more accurate representation is warranted. The staff believed that a methodology for considering the flexibility of supporting structures in the seismic design of piping systems is appropriate for the System 80+ at the certification stage. This information would be appropriate for inclusion into the DSDG. Accordingly, by DSER Open Item 3.7.3-4, the staff noted that ABB-CE should include provisions for considering equipment and support flexibility.

As previously noted, the entire DSDG was not included in CESSAR-DC. Instead, portions identified by the staff as necessary for its safety evaluations were extracted, clarified and modified as necessary and included in CESSAR-DC Appendix 3.9A (Amendment R). Section 1.7.2.8 of this appendix provides requirements for consideration of support stiffness in piping systems. ABB-CE specifies a deflection limit of 1.6 mm (1/16 in.) in the restrained direction based on the greater of the SSE load or the minimum design load in Section 1.7.2.13 of the Appendix 3.9A. In addition, the maximum deflection is limited to 3.2 mm (1/8 in.) based on the maximum load combination. This is acceptable and resolves DSER Open Item 3.7.3-4.

Interaction of Other Systems With Category I Systems

By RAI Q210.44, the staff asked ABB-CE to justify the criteria in CESSAR-DC Section 3.7.3.13 or to revise this section to be consistent with the criteria of SRP Section 3.7.3, Item II.8. ABB-CE responded to RAI Q210.44 by committing to revising CESSAR-DC Section 3.7.3.13 in accordance with SRP Section 3.7.3, Item II.8 and SRP Section 3.9.2, Item II.2.k. The draft revision submitted with the response to RAI Q210.44 only committed to designing non-Category I piping in accordance with CESSAR-DC Section 3.6.2.1.4.1.C, which relates to high-energy line-break requirements. The draft revision was not consistent with all of the applicable guidelines in SRP Sections 3.7.3 and 3.9.2 and was, therefore, unacceptable. Subsequently, by DSER Open Item 3.7.3-5, ABB-CE was asked to revise CESSAR-DC Section 3.7.3.13, as previously requested in RAI Q210.44, to be in accordance with SRP Section 3.7.3, Item II.8, and SRP Section 3.9.2, Item II.2.k.

By Amendment N, ABB-CE revised CESSAR-DC Section 3.7.3.13 to specify criteria for the protection of seismic Category I piping from possible adverse effects of other piping during earthquakes. These criteria are in accordance with SRP Section 3.9.2 Item II.2.k requirements and are, therefore, acceptable. On this basis, DSER Open Item 3.7.3-5 is resolved.

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Multiple Supported Equipment and Components With Distinct Inputs

In the early amendments to CESSAR-DC Section 3.7.3.9, ABB-CE stated that the seismic qualification of equipment or components supported at different elevations is based on the envelope of response spectra or "multiple support excitation."

By RAI Q210.41, the staff asked ABB-CE to specify that for acceptability, multiple support excitation methods should be implemented in accordance with the staff recommendations on response combinations given in NUREG-1061, Volume 4, Section 2. In its response to RAI Q210.41, ABB-CE committed to revising CESSAR-DC Section 3.7.3.9 to specify these NUREG-1061, Volume 4 recommendations and to add NUREG-1061, Volume 4 to the list of references in CESSAR-DC Section 3.7. This was DSER Confirmatory Item 3.7.3-3.

However, as discussed in Section 3.12 of this report, the staff was told that upon further review ABB-CE decided not to use the multiple support response spectrum analysis method in System 80+ piping design. In Amendment Q, this section was revised to permit only the time-history analysis method. The staff finds this revision acceptable. On this basis, DSER Confirmatory Item 3.7.3-3 is resolved.

Torsional Effects of Eccentric Masses

In the early amendments to CESSAR-DC Section 3.9.3.11, ABB-CE stated that piping system models include torsional effects due to such projecting masses as valve operators. The staff believed that ABB-CE should not limit consideration of such effects to piping systems only, but should consider them for all subsystems.

By DSER Open Item 3.7.3-6, ABB-CE was asked to revise CESSAR-DC Section 3.9.3.11 to commit to consider torsional effects of eccentric masses for all subsystems and not just for piping systems.

By Amendment N, ABB-CE revised CESSAR-DC, Section 3.7.3.11 to specify that torsional effects of eccentric masses are also considered in the analysis of seismic Category I subsystems other than piping. On this basis, DSER Open Item 3.7.3-6 is resolved.

Category I Buried Piping, Conduits, and Tunnels

In the early amendments to CESSAR-DC Section 3.7.3.12, ABB-CE stated that the intake structure would be designed so that "differential movement between this structure and

the earth is negligible." However, ABB-CE gave no indication of the magnitude of this permissible motion or the level of stresses associated with such "negligible" relative motion. By DSER Open Item 3.7.3-7, the staff noted that ABB-CE should provide generic approaches for the evaluation of the intake structure as well as acceptance criteria that will be used to evaluate that structure. This intake structure is a structural issue and the resolution of DSER Open issue 3.7.3-7 is addressed in Section 3.7.3 of this report.

In the early amendments to CESSAR-DC Section 3.7.3.12.1, ABB-CE provided design criteria for buried piping. By RAI Q210.42 and DSER Open Item 3.7.3-8, the staff asked ABB-CE to revise CESSAR-DC Section 3.7.3.12.1 to be in accordance with the criteria of SRP Section 3.7.3, Revision 2, and SRP Section 3.9.2, Revision 2. By Amendment Q and subsequent amendments, ABB-CE provided criteria in CESSAR-DC Section 3.7.3.12.1 for the design and analysis of buried seismic Category I piping systems. The detailed staff review of the buried piping issue is discussed in Section 3.12.8 of this report. On this basis, DSER Open Item 3.7.3-8 is resolved.

In the DSER, the staff noted that ABB-CE did not discuss seismic design analysis for buried or above-ground tanks and that these should be addressed in accordance with Paragraph I.14 of SRP Section 3.7.3. This issue was identified as DSER Open Item 3.7.3-9. However, these buried and above-ground tanks are structural issues and the resolution of DSER Open Item 3.7.3-9 is addressed in Section 3.7.3 of this report.

In the DSER, the staff also noted that ABB-CE should provide generic approaches to and acceptance criteria for the evaluation of buried piping, conduits, and tunnels. This was DSER Open Item 3.7.3-10. These issues as they relate to buried piping are addressed in Section 3.12.3 of this report. The structural-related issues in DSER Open Item 3.7.3-10 and their resolutions include these issues and are discussed in Section 3.7.3 of this report.

Above-Ground Piping Outside the Containment

CESSAR-DC Section 3.7.3.12.2 addresses analysis methods and criteria used to account for the effects of differential movement of buildings on piping and penetrations. In the early amendments, this section only stated that effects of differential movement of buildings on piping are described in CESSAR-DC Sections 3.7.2.1.2 and 3.7.2.7. These sections described the seismic analysis method for NSSS systems and combinations of modal responses but did not provide analysis methods and criteria for the effects of these differential motions. By RAI

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Q210.43, the staff asked ABB-CE to provide these analysis methods and criteria. Subsequently, ABB-CE submitted clarifications in its response to RAI Q210.43 and, in CESSAR-DC Section 1.2.4.1, Appendix 3.9A, committed to using the most conservative method of analysis to account for the effects of differential movement between buildings and/or equipment on piping.

Specifically, the maximum relative displacements between the major elevation and the basemat (Seismic Anchor Motions - SAMs) are to be calculated for each separately founded structure and applied conservatively (that is, out-of-phase) to piping systems which extend between adjacent separately founded structures. The staff agrees that this method of analysis is conservative and, hence, acceptable. Moreover, this method of analysis is standard industry practice and has previously been accepted by the staff on other nuclear power plants.

Reactor Internals, Core, and CEDM Seismic Subsystem Analysis

CESSAR-DC Section 3.7.3.14 provides information on the reactor internals, reactor core, and CEDM seismic subsystem analysis.

As stated in the CESSAR-DC, the seismic analysis of the reactor internals and core consists of a preliminary phase and a final phase. The preliminary phase is used to obtain preliminary seismic design loads and displacements in the vertical and horizontal directions and the final phase to confirm that the design-basis loads determined during the preliminary phase are conservative and result in an adequate design.

Separate analyses are performed in the vertical and two horizontal directions. In the horizontal direction, because of the closure of gaps resulting from relative component displacements, a nonlinear horizontal time-history analysis is performed. In the vertical direction, the preliminary analysis used the linear modal analysis response spectrum method. However, if the preliminary linear vertical analysis indicates that the response of the core may cause the core plate to lift off, a vertical nonlinear analysis is also performed.

The maximum responses in the two horizontal components and one vertical component are considered and the maximum colinear responses from each are combined by the SRSS method.

As described, the modeling procedures, methods of analysis, procedures for damping, the combination of modal

responses, and the three components of earthquake motion are in accordance with the criteria of SRP Section 3.7.3 and, hence, are acceptable.

CESSAR-DC Section 3.7.3.14.2 states that structural integrity of the CEDM, when subjected to seismic loadings, is verified by a combination of test and analysis. The analysis uses either the response spectrum or time-history method. A functional test using a minimum drop weight is performed and the results are compared to the calculated CEDM deflections under seismic loadings for the individual site. However, since the System 80+ CEDM design is identical to the design in operation at Palo Verde, as stated in CESSAR-DC Section 3.9.4.4.1.3, and since the System 80+ plant is designed to an enveloped seismic input, the CEDM seismic analysis can be performed for the System 80+ CEDM design and those results can be compared to the drop test results that were performed for the System 80 CEDM design in use at Palo Verde.

By DSER Open Item 3.7.3-11, ABB-CE was asked to perform the seismic analysis for the CEDM design using the enveloped seismic input for the System 80+ plant and then compare those results to the results of the drop tests that were performed for the System 80 plant design, as described in CESSAR-DC Section 3.7.3.14.2.

Subsequently, during the June 23 through 25, 1993, audit the staff reviewed ABB-CE's analysis and evaluation results. The analysis evaluated the System 80+ CEDM under seismic loadings. The evaluation included verification of operability and assessment of seismic loads. The soil case analyzed was soil Case B-4 and control motion CMS-2. The calculation documented that the input response spectrum for this combination had the highest acceleration at 20 Hz, which is the frequency of the CEDM second mode of vibration. This second mode is the most critical mode for assessing CEDM operability under seismic loadings. Accordingly, the use of soil Case B-4 and control motion CMS-2 is acceptable.

The evaluation showed that the System 80+ displacement limits and radii of curvature under load were less severe than those for the Palo Verde drop tests that were done to verify the acceptability of the CEDMs for Palo Verde. On this basis, DSER Open Item 3.7.3-11 is resolved.

On the basis of its review of CESSAR-DC Sections 3.7.3 and 3.9.2.2, the results of which are set forth above, the staff concludes that ABB-CE has met the relevant requirements of GDC 2 with respect to demonstrating design adequacy of all seismic Category I mechanical systems, equipment and their supports to withstand earthquakes by

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meeting RGs 1.61 and 1.92 guidance and by providing acceptable seismic system analysis procedures and criteria.

CESSAR-DC Appendix 3.7A: Coupled RCS Seismic Results

CESSAR-DC Appendix 3.7A (two tables) presents seismic analysis results of the RCS. CESSAR-DC Table 3.7A-1 lists dominant frequency and mode shape information for 23 significant frequencies. In the early amendments, the table did not specify the soil or RCS support conditions associated with the tabulated results. CESSAR-DC Table 3.7A-2 lists the SSE loads and design specification loads at support locations in the RCS. In the early amendments, this table also did not identify the soil or RCS support conditions for the tabulated results, nor did it explain the significance of the tabulated design specification loads.

In response to a staff request (Q220.28) for a discussion of how the design specification data for the RCS were obtained (LD-92-0166, February 12, 1992), ABB-CE stated that these seismic loads were based on past experience and were conservative enough for the design of the System 80+ RCS so that in no case would the maximum calculated values exceed these design specification seismic loads. ABB-CE committed to revising CESSAR-DC Table 3.7A-2 to provide additional clarification for staff review.

By DSER Open Item 3.7.3-12, ABB-CE was asked to (1) clarify the relationship between the results provided and the soil or RCS support conditions; (2) explain the design specification loads; and (3) expand results of Table 3.7A-1 to include RCS piping information to enable evaluation of RCS component interaction effects.

Moreover, DSER Section 3.7.3 noted that, in addition to the information presented in then current CESSAR-DC Appendix 3.7A, other information that was needed to analyze subsystems (e.g., SL) attached to the RCS should be presented. This information was to include, but not limited to, the accelerations, displacements, and FRS at appropriate locations. The same information for all seismic Category I buildings or structures must also be presented in CESSAR-DC Section 3.7.

The information provided in CESSAR-DC Appendix 3.7A and requested additional information described in the preceding is typical of analysis results usually provided in FSARs for plants licensed under 10 CFR Part 50. However, in view of the DAC approach for the standard System 80+ design, such results are considered as for information only but do provide some assurance that the plant can be reasonably designed.

Accordingly, the expansion of Table 3.7A-1 to include additional RCS piping information requested in DSER Open Item 3.7.3-12 and other information identified in DSER Section 3.7.3 as needed to analyze subsystems were determined to be not necessary at this time.

By Amendment N to CESSAR-DC, Tables 3.7A-1 and 3.7A-2 were modified. Information for 23 significant frequencies was provided in Table 3.7A-1 and the design specification loads were deleted from Table 3.7A-2. In addition, ABB-CE provided, in Amendment T to CESSAR-DC, an overview section of Appendix 3.7A which clarified that Table 3.7A-1 was based on rigid coupling of the RCS to the building and Table 3.7A-2 was based on the envelope of all SSE soil cases. These clarifications are acceptable. On this basis, DSER Open Item 3.7.3-12 is resolved.

3.9.2.3 Dynamic System Analysis Methods for Reactor Vessel Core Support and Internal Structures

GDC 1 and 4 require that structures and components important to safety (such as reactor vessel core support and internal structures) be constructed and tested to quality standard commensurate with the importance of the safety functions to be performed, and designed with appropriate margins to withstand dynamic effects of anticipated normal plant occurrences, natural phenomena such as earthquakes, and postulated accidents including LOCA. In accordance with SRP Section 3.9.2, Item II.3, analytical solutions to predict vibrations of reactor internals are required for prototype plants.

CESSAR-DC Section 3.9.2.3 describes the flow-induced vibration analysis of the reactor internal components during normal operating conditions. Both deterministic (periodic and transient) and random pressure fluctuations in the coolant are considered.

CESSAR-DC Section 3.9.2.4 states that the System 80+ reactor structures are designated as a non-prototype Category I (per RG 1.20) design, and that the Palo Verde, Unit 1, System 80 reactor was the valid prototype. CESSAR-DC Section 3.9.2.4 also states that it has documented correlations between analytical predictions and test results for dynamic responses of the reactors ISs during normal operating conditions for the Palo Verde, Unit 1 plant in, "A Comprehensive Vibration Assessment Program for PVNGS Unit 1 (System 80 Prototype)," CEN-263, Revision 1, January 1985. By RAI Q210.53, the staff asked ABB-CE to compare the Palo Verde (Unit 1) and the System 80+ designs in accordance with the criteria in SRP Section 3.9.2, Revision 2, to support the designation of Palo Verde as the prototype for the Sys-

tem 80+ design. ABB-CE responded to RAI Q210.53 by stating that the designs as described in the CESSAR-DC and the Palo Verde (Unit 1) final safety evaluation report are essentially the same and concluded that the dynamic characteristics of the designs would be nearly identical. This response was not totally acceptable. Accordingly, ABB-CE was asked in DSER Open Item 3.9.2.3-1 to submit information to validate the designation of Palo Verde (Unit 1) as the prototype for the System 80+ design. Subsequently, CESSAR-DC Tables 1.3-1, 3.9-17, and 3.9-18 compared the System 80+ and the Palo Verde (Unit 1) designs, differences between the designs and the reactor vessel internal parameters that demonstrate that the design differences will not substantially alter the behavior of the flow transients, and noted the response of the reactor internals. On this basis, DSER Open Item 3.9.2.3-1 is resolved.

For random force input analyses, ABB-CE stated in CESSAR-DC Section 3.9.2.3.5.2 that only root-mean-square (RMS) displacements, loads, and stresses are calculated. In RAI Q210.52, the staff asked ABB-CE to commit to including methods for considering peak responses to random force input-analyses. ABB-CE responded to RAI Q210.52 by stating that three times the RMS response results from random force input-analyses are combined with similar response results from deterministic analyses and the combined response used in design verification analyses. By Amendment U to CESSAR-DC, ABB-CE further stated that the use of three times the RMS values was common design practice for random Gaussian loadings. This practice may prove potentially unconservative for loadings with very large numbers of cycles. However, this potential unconservatism is more than offset by ABB-CE's commitment in CESSAR-DC Section 3.9.2.3.5.2 to combine three times the RMS values with results from other analyses (i.e., deterministic thermal, etc.). On this basis, DSER Open Item 3.9.2.3-2 is resolved.

The System 80+ design of reactor internals meets the relevant requirements of GDC 1 and 4 with respect to the reactor internals being designed and tested to quality standards commensurate with the importance of safety functions being performed and being appropriately protected against dynamic effects by meeting the regulatory positions of RG 1.20 for the conduct of preoperational vibrational tests and by having a preoperational vibration program planned for the reactor internals that provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provides adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced

vibrations of reactor operation without loss of structural integrity.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

CESSAR-DC Section 3.9.2.5 provides information on dynamic analyses intended to confirm the structural adequacy and ability, with no loss of function, of the reactor internals and unbroken loops of the reactor coolant piping to withstand the loads from a LOCA in combination with the SSE. In the early CESSAR-DC amendments, ABB-CE did not evaluate the faulted condition for the reactor internals and the unbroken loops of the reactor coolant piping in accordance with the criteria of SRP Section 3.9.2, Revision 2, Item II.5. Specifically, the effects of postulated ruptures in accordance with SRP Section 3.6.2, Revision 2, was not included in the analysis. However, on the basis of NRC approval (in Section 3.6.3 of this report), of the application of the LBB methodology to the System 80+ design, ABB-CE need not include the effects of pipe ruptures eliminated by the LBB in the faulted condition. CESSAR-DC Section 3.9.2.5 states that the faulted condition is evaluated on the basis of 110 percent of the SSE loading on the assumption that a 10 percent increase of the SSE is sufficient to account for RCS branch line pipe breaks not eliminated by the LBB methodologies. ABB-CE did calculations to verify that the additional 10 percent of the SSE loading would envelope the effects of RCS branch line breaks not eliminated by the LBB methods. On the basis of its audit review of the calculation results, the staff concludes that the 110 percent of SSE effects provides an upper bound for the effects due to any uneliminated branch line breaks on the reactor internals.

By DSER Open Item 3.9.2.4-1, the staff also noted that no results of the effect from RCS branch line break were provided for unbroken loops of MCL piping. ABB-CE responded in a letter dated December 18, 1992 that CESSAR-DC Section 3.9.2.5 was not intended to address evaluation of RCS unbroken loops under faulted conditions but only to address the load effect of RCS branch line breaks to reactor internals. Furthermore, ABB-CE stated that following the application of LBB to System 80+ design, the largest primary side pipe break that would load the RCS unbroken loops was the 76-mm (3-in.) spray-line break. The loading effects of this break on MCL piping with outside diameters of 0.91 m and 1.25 m (36 and 49 inches) would be insignificant. On this basis, DSER Open Item 3.9.2.4-1 is resolved.

In summary, on the basis of its review of CESSAR-DC Sections 3.9.2.4 and 3.9.2.5 as described above, the staff concludes that ABB-CE meets the relevant requirements of

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GDC 2 and 4 with respect to the design of systems and components important to safety to withstand the effects of earthquakes and the appropriate combinations of the effects of normal and postulated accident conditions with the effects of the SSE by performing dynamic system analyses and demonstrating that the structural design of the reactor internals and unbroken piping loops are adequate to withstand the combined loads of postulated LOCAs and the SSE.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core-Support Structures

GDC 1, 2, and 4 and 10 CFR 50.55a require that structures and components important to safety be constructed and tested to quality standard commensurate with the importance of the safety functions to be performed, and designed with appropriate margins to withstand effects of anticipated normal plant occurrences, natural phenomena such as earthquakes and postulated accidents including LOCA. GDC 14 and 15 require that the reactor coolant pressure boundary design limits will not be exceeded.

The staff's review under SRP Section 3.9.3 concerns the structural integrity and functional capability of pressure-retaining components, their supports, and core-support structures that are designed in accordance with ASME Code, Section III, Division 1. The staff reviewed loading combinations and their respective stress limits, the design and installation of pressure-relief devices, and the design and structural integrity of ASME Code, Class 1, 2 and 3 components and component supports, and Class CS core-support structures.

By DSER Open Item 3.9.3-1, ABB-CE was asked to submit the DSDG in its entirety for staff review. Subsequently, ABB-CE transmitted a final DSDG dated January 15, 1993 to the NRC for review. This is acceptable and resolves Open Item 3.9.3-1. Section 7 of the DSDG gives detailed requirements for the design of piping, HVAC components, and cable-tray/conduit systems and supports. The staff did an audit review of the DSDG and noted information that needs to be incorporated into CESSAR-DC Appendix 3.9A. The staff's evaluation of the DSDG is reported in Section 3.12 of this report. The staff reports its review of cable-tray, conduit, and HVAC support systems in Section 3.7.3 of this report.

3.9.3.1 Loading Combination, System Operating Transients, and Stress Limits

CESSAR-DC Section 3.9.3.1 identifies the load combinations and stress limits to be used in the design of ASME Code, Section III, Class 1, 2, and 3 components and

component supports and Class CS core-support structures. The staff's review of CESSAR-DC Section 3.9.3.1, in accordance with SRP Section 3.9.3, Revision 1, is as follows.

In the early amendments of CESSAR-DC Section 3.9.3, ABB-CE provided a number of tables that specify loading combinations for ASME Code, Section III components. Specific sustained loads found in the loading combinations in these tables for the various service levels defined in ASME Code were incomplete or in need of clarification. For example, for Levels A, B, C, and D service conditions, except for transients and the OBE and SSE, the only other load specified was deadweight. Pressure, thermal expansion, and differential seismic and thermal anchor movement loads were not specified in the loading combinations. In response to DSER Open Item 3.9.3.1-1 and staff comments in the meeting of June 23-25, 1993, ABB-CE revised CESSAR-DC Tables 3.9-10 through 3.9-14, in Amendment Q, to specify normal operating pressure, thermal expansion, and differential seismic anchor movement loads for various load combinations. On this basis, DSER Open Item 3.9.3.1-1 is resolved.

In a number of loading combination tables in CESSAR-DC Section 3.9.3 (Amendment E), ABB-CE stated that for ASME Code, Section III, Class 1, 2, and 3 components, including piping, and for component supports, Level C and D transients were included in the Level C and D loading combinations, respectively, but these transients were not in the loading tables. In response to DSER Open Item 3.9.3.1-2, ABB-CE revised the CESSAR-DC (Amendment Q), and noted applicable Level C and D transient events in CESSAR-DC Tables 3.9-10 through 3.9-12. On this basis, DSER Open Item 3.9.3.1-2 is resolved.

By RAI Q210.55, the staff asked ABB-CE to revise all loading combination tables to clearly state that the Level D loading combination includes LOCA loads. Contrary to this request, ABB-CE responded to RAI Q210.55 by committing to revise load combination Tables 3.9-2, 3.9-10, 3.9-11, and 3.9-12 to include loads due only to postulated "branch line" pipe breaks (LOCA and secondary side) not eliminated by LBB in the level D loading combinations. As discussed in Section 3.6.3 of this report, it is acceptable for the loading combination tables not to include the loads due to pipe ruptures that are eliminated on the basis of LBB. On this basis, DSER Open Item 3.9.3.1-3 is resolved.

In the early amendments to CESSAR-DC Section 3.9.3, a number of loading combination tables indicated that dynamic loads were combined by the SRSS method. By RAI Q210.57, the staff asked ABB-CE to modify

CESSAR-DC Section 3.9.3 and the loading combination tables to show that the loads will be combined in accordance with the guidelines of NUREG-0484, Revision 1 ("Methodology for Combining Dynamic Responses," U.S. NRC, May 1980). According to NUREG-0484, the staff considers the SRSS method of combining dynamic responses appropriate for dynamic responses resulting from SSE and LOCA; for the same initiating event, when the time-phase relationship of the responses can be established; and for dynamic responses other than SSE or LOCA, provided that a non-exceedance probability of 84 percent or higher is achieved for the combined SRSS response. By DSER Confirmatory Item 3.9.3.1-1, ABB-CE was asked to commit to complying with the NUREG-0484 guidelines. In Amendment K to CESSAR-DC, ABB-CE revised Section 3.9.3.1 to state that the loads will be combined in accordance with the guidelines of NUREG-0484. On this basis, DSER Confirmatory Item 3.9.3.1-1 is resolved.

By RAI Q210.58 and part of DSER Open Item 3.9.3.1-4, the staff requested that for all ASME Code, Section III, Class 2 and 3 components, equipment, and supports that are designed for a 60-year life and subject to loadings that would result in thermal or dynamic fatigue, ABB-CE should commit to perform a cyclic evaluation similar to those required for Class 1 components in Subsection NB of the ASME Code, Section III. In response, ABB-CE revised CESSAR-DC Section 3.9.3.1.3 (Amendment R) to state that Class 2 and 3 components are reviewed for thermal fatigue effects using the ASME Code, Section III, NC-3219.2, for guidance and that fatigue analysis is performed in accordance with NC-3200 for these components if required. The staff finds that the commitment for fatigue evaluation of Class 2 and 3 components is in accordance with ASME Code, Section III, Appendices XIII and XIV, and is, therefore, acceptable. This resolves the applicable part of DSER Open Item 3.9.3.1-4. In the same DSER open item, the staff also raised a concern about environmental effects on the fatigue life for Class 2 and 3 components. ABB-CE, in CESSAR-DC Section 3.9.1.1, stated that environmental degradation had not been identified as a significant issue to the System 80+ design because System 80+ is not exposed to high-oxygen-content environments at elevated temperatures, and because no carbon or low alloy steel is directly exposed to the primary coolant. This clarification is found acceptable, as discussed in Section 3.9.1 of this report. On this basis, the staff concludes that, unless new evidences is developed, the System 80+ design need not consider environmental effects in the fatigue analysis for Class 2 and 3 components, equipment, and supports.

By RAI Q210.90 the staff requested, in part, that ABB-CE include the effects of thermal stratification in the design of the SL. ABB-CE responded to RAI Q210.90 by commit-

ting to include thermal stratification and thermal striping effects in the SL analysis. More generally, by NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," the staff is requesting that licensees and applicants review systems connected to the RCS to determine whether any sections of such piping that cannot be isolated can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves. By DSER Open Item 3.9.3.1-5, ABB-CE was asked to conform to the guidelines of NRC Bulletins 88-08 and 88-11 for piping systems connected to the RCS. ABB-CE addressed these thermal stratification issues in Section 1.4.7 of CESSAR-DC Appendix 3.9A (Amendment R). The staff's detailed evaluation of these issues is reported in Sections 3.12.5.9 and 3.12.5.10 of this report.

CESSAR-DC Appendix 3.9A (Section 1.4.7) describes the program that was established by the owners and operators of ABB-CE plants to respond to Bulletin 88-08. The results of the program determined that thermal stratification can exist in the DVI line for System 80+. The CESSAR-DC states that available data from the program are used to evaluate the thermal stresses in DVI lines that occur because of thermal stratification. In addition, the System 80+ SC lines are arranged in a similar fashion to the arrangements in the ABB-CE's operating plants. The CESSAR-DC notes that measurements on such lines obtained to date from operating reactors are inconclusive. As a result, ABB-CE states that conclusions from new data evaluations will be incorporated into the design of the System 80+ SC lines. Furthermore, ABB-CE also evaluates the location and orientation of the SDS and concludes that thermal stratification should not occur in the System 80+ SDS lines. On this basis, the staff concludes that ABB-CE adequately addresses the potential problems described in Bulletin 88-08.

The CESSAR-DC also describes the program that was sponsored by the owners and operators of ABB-CE operating plants to address Bulletin 88-11. That program obtained data at four operating plants which showed that the maximum temperature differences in the SL were bounded by the difference in temperature between the pressurizer and the hot leg. ABB-CE states that the System 80+ SL is designed for this maximum temperature difference and gives detailed evaluation in CESSAR-DC Sections 3.12.5.9 and 3.12.5.10. This commitment conforms to Bulletin 88-11 and is acceptable. On this basis, DSER Open Item 3.9.3.1-5 is resolved.

In Amendment E to CESSAR-DC Section 3.9.3.1.4.2, Item C (functional capability), ABB-CE stated that the criteria in Texas Utilities Electric Co. letter TXX-3423 would be used to satisfy functional capability requirements

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in piping systems and that the NRC reviewed and accepted the criteria of TXX-3423. By DSER Open Item 3.9.3.1-6, the staff noted that these criteria were only applicable to stainless steel elbows and that ABB-CE should also provide criteria for other piping products and joints and other materials. In response to DSER Open Item 3.9.3.1-6 and the staff's DSDG review comments, ABB-CE included a new CESSAR-DC Section 3.9.3.1.4.3 for Class 1, 2, and 3 piping systems (Amendment Q) to state that the piping is designed to meet a D/t ratio of less than 50, in accordance with NUREG-1367 ("Functional Capability of Piping Systems," U.S. NRC, November 1992), and is, therefore, acceptable. The objective of this NUREG report is to determine whether present code rules and potential changes to the code rules are sufficient to ensure maintenance of functional capability. More detailed discussion of the functional capability requirements and the objective of NUREG-1367 is provided in Section 3.12.5.12 of this report. On this basis, DSER Open Item 3.9.3.1-6 is resolved.

Per SRP Section 3.9.3, Items I.1 and II.1, the staff reviews loading combinations and stress limits for internal parts of components such as valve discs and seats, and pump shafting subjected to dynamic loading during operation of the components. By RAI Q210.63, the staff asked ABB-CE to provide design criteria for these internal parts in CESSAR-DC Section 3.9.3. ABB-CE responded to RAI Q210.63 by stating that CESSAR-DC Section 3.9.3.2, "Pumps and Valve Operability Assurance," covers the structural integrity of the pumps' and valves' internal parts. However, ABB-CE revised CESSAR-DC Sections 3.9.3.1 and 3.9.3.2.1.1.A (Amendment N) to explicitly include internal parts of components. CESSAR-DC Tables 3.9-7 and 3.9-9 for active pumps and valves limit the stresses to 1.2 S and 1.8 S for membrane and membrane plus bending, respectively. Implementation of such criteria will provide assurance that the maximum distortions of the parts are within their deflection limits so as not to impair the operability of the equipment. This is acceptable. On this basis, DSER Open Item 3.9.3.1-7 is resolved.

By RAI Q210.67, the staff noted that the ASME Code requires that a design specification be prepared for all ASME Class 1, 2, and 3 components such as pumps, valves and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations and other design data. The code also requires a design report for all such components. However, most, if not all, Class 1, 2, and 3 pump, valve, and piping system design reports are provided by vendors and would not be available at the time of design certification. Therefore, in response to DSER Open

Item 3.9.3.1-8, ABB-CE developed a procedure guideline for preparing ASME Code, Section III design specifications, which will serve as the basis for construction (as defined by ASME Code, Section III, Subsection NCA-1110) of components necessary for a vendor to prepare a design report. In CESSAR-DC Section 3.9.3.1.5 (Amendment Q), ABB-CE stated that the cited guidelines are used to prepare the design specifications. This is acceptable and resolves DSER Open Item 3.9.3.1-8.

CESSAR-DC Section 9.4, "Air Conditioning, Heating, Cooling and Ventilation Systems," describes the ventilation systems for the different plant buildings and areas. For each system, certain codes, standards, and practices (e.g., Sheet Metal and Air Conditioning Contractors National Association (SMACNA); ASME/ANSI AG-1; and ASME Code, Section III, Class 3) are referenced as being applicable to the manufacture, installation, testing, and performance of HVAC equipment. By RAI Q210.68, the staff asked ABB-CE to describe in detail how these codes, standards, and practices address the seismic design criteria, installation criteria, and QA requirements for safety-related HVAC equipment, including ductwork, duct supports, cooling and heating coils, piping, and piping supports.

ABB-CE responded to RAI Q210.68 by stating that the codes and standards that are referenced in CESSAR-DC Section 9.4 do not specifically detail the criteria for seismic design, installation, or QA for safety-related HVAC equipment. The ductwork is seismically designed in accordance with ASME/ANSI AG-1-1988, Article AA-4000, "Structural Design." The cooling and heating coils, piping, and pipe supports are seismically designed in accordance with ASME Code, Section III, Class 3. Since it is primarily a structural issue, the staff reports its review of cable tray, conduit, ductworks, and HVAC in Section 3.7.3 of this report. By DSER Open Item 3.9.3.1-9, the staff asked ABB-CE to submit explicit information regarding the design for constructing duct supports. DSER Open Item 3.9.3.1-9 is primarily a structural issue. The adequacy of the duct and duct support design is determined by structural review. On this basis, DSER Open Item 3.9.3.1-9 is resolved as a mechanical issue.

On the basis of its review of CESSAR-DC Section 3.9.3.1, the staff finds that ABB-CE meets 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code, Class 1, 2, and 3, components. It meets 10 CFR 50.55a and GDC 1, 2, and 4 by ensuring that (1) systems and components important to safety are designed to quality standards commensurate with their importance to safety and (2) these systems and components can accommodate the effects of NOP as well

as such postulated events as LOCAs and the dynamic effects resulting from earthquakes. The specified design and service combinations of loading as applied to ASME Code, Class 1, 2, and 3 pressure-retaining components in systems designed to meet seismic Category I standards provide assurance that, in the event of an earthquake affecting the site or other service loading caused by postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting stresses under such loading combinations provides a conservative design basis for ensuring that the system components will withstand the most adverse combination of loading events without loss of structural integrity.

3.9.3.1.1 Intersystem LOCA Design for Piping Systems

In SECY-90-016 dated January 12, 1990, the NRC staff recommended that the Commission approve the staff's resolution of the intersystem LOCA (ISLOCA) issue for advanced LWR plants by requiring that low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. As noted in its June 26, 1990 SRM, the Commission approved the staff's recommendation provided that all elements of the low-pressure system are considered. ABB-CE provided its proposed implementation of the issue resolution for the System 80+ design in CESSAR-DC Appendix 5E. ABB-CE proposed, in part, in CESSAR-DC Appendix 5E, Section 3.3, to design the interfacing systems and subsystems to at least 40 percent of the RCS normal operating pressure.

The staff evaluated the ABB-CE proposal for implementing the ISLOCA resolution for the System 80+ design. As an applicable regulation described in Section 1.6 of this report, the staff proposed that

the standard design must minimize the effects of intersystem loss-of-coolant accidents by designing low-pressure piping systems that interface with the RCPB to withstand full reactor coolant system pressure to the extent practical.

In Section 20.2 of this report, under Issue 105, the staff evaluated ABB-CE's approach, in terms of the practicality for systems, components, and equipment, for implementing the ISLOCA resolution for the System 80+. In the following, the staff evaluated the minimum pressure for which low-pressure systems should be designed to ensure reasonable protection against burst failure should the low-pressure system be subjected to full RCS pressure. In establishing the minimum design pressure, the following goals were used as the basis for selection:

- (1) The likelihood of rupture (burst) of the pressure boundary is low, based on the staff's goal of approximately 10-percent component failure probability for rupture.
- (2) The likelihood of intolerable leakage of flange joints or valve bonnets is reasonably low although some leakage might occur.
- (3) Some piping components might undergo gross yielding and permanent deformation.

3.9.3.1.1.1 Low-Pressure Piping Design

To achieve its objectives, the staff evaluated, first, on a qualitative basis, several possible ratios of the low-pressure system design pressure (P_d) to the RCS normal operating pressure (P_v) to establish the margins on burst and yield of the piping. The results of the staff's evaluation are depicted in Table 3.3 of this chapter for typical carbon steel (SA-106 Grade B) and stainless steel (SA-312 Types 304 and 316) material and are discussed below for three ratios of the design pressure to the reactor vessel pressure (P_d/P_v). A margin of 1.0 or less represents the condition where burst or yielding is likely to occur. The higher the margin, the less likely it is for burst or yielding to occur. The low-pressure piping systems are assumed to be designed to the rules of the ASME Code, Section III, Subarticle NC/ND-3600 for Class 2 and 3 piping systems.

Piping Integrity at $P_d/P_v = 1/2$ (ASME Code Service Level D)

When P_d/P_v is equal to one-half, the margins on burst and yield are equivalent to approximately those of the ASME Boiler and Pressure

Vessel Code, Section III, Service Level D condition. For carbon steel pipe (SA-106 Grade B material), this ratio will provide a margin of 2.0 on burst and 1.08 on yield for a pipe at 260 °C (500 °F). For stainless steel piping, a ratio of one-half will provide a sufficient margin on burst (1.7 and 1.65 for SA-312 Type 304 and SA-312 Type 316 materials, respectively). However, a small amount of yielding is likely to occur with a margin of 0.70 for both stainless steels at 260 °C (500 °F). No leakage of the pressure boundary is likely to occur at P_d/P_v equal to one-half.

As a result, a ratio of one-half will ensure the pressure integrity of the low-pressure piping system with ample margin.

Table 3.3 Margins for straight pipe

Material	Temp °C (°F)	P_d/P_v	S MPa (ksi)	S_v MPa (ksi)	S_u MPa (ksi)	S_y MPa (ksi)	Margins on Burst Yield	
SA-106 Grade B	37.8 (100)	1/2	103.4 (15)	206.8 (30)	413.7 (60)	241.3 (35.0)	2.00	1.34
	260 (500)	1/2	103.4 (15)	206.8 (30)	413.7 (60)	195.1 (28.3)	2.00	1.08
	37.8 (100)	1/3	103.4 (15)	310.3 (45)	413.7 (60)	241.3 (35.0)	1.33	0.89
	260 (500)	1/3	103.4 (15)	310.3 (45)	413.7 (60)	195.1 (28.3)	1.33	0.72
	37.8 (100)	1/4	103.4 (15)	413.7 (60)	413.7 (60)	241.3 (35.0)	1.00	0.67
	260 (500)	1/4	103.4 (15)	413.7 (60)	413.7 (60)	195.1 (28.3)	1.00	0.54
SA-312 Type 304	37.8 (100)	1/2	129.6 (18.8)	258.6 (37.5)	517.1 (75.0)	206.8 (30.0)	1.70	0.92
	260 (500)	1/2	109.6 (15.9)	219.3 (31.8)	437.8 (63.5)	133.8 (19.4)	1.70	0.70
	37.8 (100)	1/3	129.6 (18.8)	388.2 (56.3)	517.1 (75.0)	206.8 (30.0)	1.13	0.61
	260 (500)	1/3	109.6 (15.9)	328.9 (47.7)	437.8 (63.5)	133.8 (19.4)	1.13	0.47
	37.8 (100)	1/4	129.6 (18.8)	517.1 (75.0)	517.1 (75.0)	206.8 (30.0)	0.85	0.46
	260 (500)	1/4	109.6 (15.9)	438.5 (63.5)	437.8 (63.5)	133.8 (19.4)	0.85	0.35
SA-312 Type 316	37.8 (100)	1/2	129.6 (18.8)	258.6 (37.5)	482.6 (70.0)	206.8 (30.0)	1.59	0.92
	260 (500)	1/2	109.6 (15.9)	219.3 (31.8)	424.7 (61.6)	137.2 (19.9)	1.65	0.72
	37.8 (100)	1/3	129.6 (18.8)	388.2 (56.3)	482.6 (70.0)	206.8 (30.0)	1.06	0.61
	260 (500)	1/3	109.6 (15.9)	328.9 (47.7)	424.7 (61.6)	137.2 (19.9)	1.10	0.48
	37.8 (100)	1/4	129.6 (18.8)	517.1 (75.0)	482.6 (70.0)	206.8 (30.0)	0.80	0.46
	260 (500)	1/4	109.6 (15.9)	438.5 (63.6)	424.7 (61.6)	137.2 (19.9)	0.85	0.36

S = allowable stress per ASME Code, Section III for Class 2 piping

S_v = hoop stress at $P = P_v$
 $= S/(P_d/P_v)$

S_u = ultimate tensile strength; from Section III, Table 1-3.1 and 1-3.2

S_y = yield strength; from Section III, Table 1-2.1 and 1-2.2

Margin on burst pressure = $F \times S_u \times (P_d/P_v)/S$

where $F = 1.00$ for SA-106 Grade B

$F = 0.85$ for SA-312 Type 304 & Type 316

Margin on yield pressure = $1.15 \times S_y \times (P_d/P_v)/S$

Piping Integrity at $P_d/P_v = 1/3$

When P_d/P_v is reduced to one-third, the margins for carbon steel piping (SA-106 Grade B material) are lowered to 1.33 and 0.72 for burst and yield at 260 °C (500 °F), respectively. For stainless steel piping, the margins are 1.13 and 0.47 for burst and yield at 260 °C (500 °F), respectively for SA-312 Type 304 material and 1.10 and 0.48 for burst and yield, respectively, for SA-312 Type 316 material. At these margins, it is expected that burst failure will not occur in either carbon steel or stainless steel piping. However, a significant amount of yielding might occur in stainless steel piping at all temperatures and in carbon steel piping at 260 °C (500 °F). Where the carbon steel piping is at a lower temperature, some yielding might occur although to a lesser extent. The consequence of significant pipe yielding (without bursting) is that gross, permanent distortion might occur in the piping components thereby resulting in some leakage through flanges, or valve bonnets. However, it is not expected that such leakage would be uncontrollable or intolerable.

In summary, a ratio of one-third will ensure the pressure boundary of the low-pressure piping although a significant amount of pipe yielding and some leakage through flanges and valve bonnets is likely to occur.

Piping Integrity at $P_d/P_v = 1/4$

At P_d/P_v equal to one-fourth, the pressure integrity of carbon steel piping becomes questionable, and for stainless steel piping, it is likely that burst failure will occur. Prior to bursting, the piping system would undergo gross plastic deformation, experience a significant amount of leakage at flanges, valve bonnets, and pump seals, and possibly lose some pipe supports due to the radial expansion of the pipe.

Therefore, at P_d/P_v equal to one-fourth, the ability of the low-pressure piping system to withstand full RCS pressure is questionable for carbon steel piping and unlikely for stainless steel piping systems.

The staff further evaluated, on a quantitative basis, the survival probabilities of the low-pressure piping at various design pressures using the methodology described in NUREG/CR-5603, "Pressure-Dependent Fragilities for Piping Components," dated October 1990. Calculations were performed by Idaho National Engineering Laboratory (INEL) under contract with the NRC's Office of Nuclear Regulatory Research.

The INEL calculations led to results similar to the qualitative conclusions discussed above. A temperature of 177 °C (350 °F) was used in the calculations of the following survival probabilities. Using a temperature of 260 °C (500 °F), the survival probabilities decrease about 2-5 percent for the different materials and design pressures.

For carbon steel piping (SA-106 Grade B material) with wall thickness equal to the minimum thickness required by the ASME Code for 40 percent of RCS normal operating pressure, that is, a pressure of 6.21 MPa (900 psig) (or approximately $P_d/P_v = 0.4$), the survival probability is 99 percent. For stainless steel piping (SA-312 Types 304 and 316 materials), the survival probability at 6.21 MPa (900 psig) (or approximately $P_d/P_v = 0.4$), was less than 85 percent.

These survival probabilities are based on the minimum wall thickness calculated using Eq. (3) in the ASME Code, Section III, Subarticle NC/ND-3640. The wall thickness thus calculated does not account for manufacturing tolerances or the use of the next heavier, commercially available wall thickness, which would increase the piping wall thickness and also increase the survival probability. Increasing the wall thickness to the minimum commercially available thickness required to satisfy the ASME Code minimum required thickness results in minimum survival probabilities of greater than 99, greater than 87, and less than 85 percent for SA-106 Grade B, SA-312 Type 304, and SA-312 Type 316 material respectively. On this basis, the staff found that the ABB-CE approach to designing the interfacing systems and subsystems to 40 percent of the RCS normal operating pressure would not attain the 90-percent survival probability goal in the case of stainless steel systems and subsystems.

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Subsequently the staff determined that if the wall thickness of stainless steel piping systems is of standard weight in piping of diameter 35.6 cm (14 in.) and less and is Schedule 40 in piping of diameter 40.6 cm (16 in.) and greater, the 90-percent survival probability goal will be attained. The minimum probabilities for Type 304 and Type 316 material were 92.7 and 87.2 percent, respectively. The staff discussed these results with ABB-CE and ABB-CE committed to modifying their option A design approach to also specify these minimum wall thicknesses for both carbon and stainless steel piping to achieve the 90-percent survival probability goal. This was DSER Confirmatory Item 3.9.3.1-1. For carbon steel piping, the commitment to the 40 percent of RCS normal operating pressure alone will achieve the 90-percent goal. However, for stainless steel piping, the wall thicknesses based on this design pressure will be less than those required to attain the 90-percent survival probability goal. Accordingly, the extension of the minimum 40-percent design pressure and the minimum wall thicknesses to both carbon and stainless steel low-pressure piping systems will attain the 90-percent goal. These minimum pressures for the low-pressure piping systems are identified as certified design commitments for the System 80+ design.

Valves in Low-Pressure Systems

For the valves in the low-pressure piping systems (excluding the pressure isolation valves which are already designed for RCS pressure), the selection of the valve class rating is a primary factor for designing against full RCS pressure. For example, ANSI B16.34 valves are supposed to be shop-tested to 1.5 times their 37.8 °C (100 °F) rated pressure. This would mean that for a Class 600 A216 WCB (cast carbon steel) valve, the test pressure is $1.5 \times 10.2 = 15.3$ MPa ($1.5 \times 1480 = 2220$ psig). For Class 900, the valve test pressure is $1.5 \times 15.3 = 230$ MPa ($1.5 \times 2220 = 3330$ psig).

The Class 600 valve that is tested to a pressure of 15.3 MPa (2220 psig) would be expected to withstand an RCS normal operating pressure of 15.4 MPa (2235 psig). However, it should not be assumed that the valve in the low-pressure system would be able to operate with this full RCS pressure across the disk.

Therefore, the staff finds that a Class 600 valve is adequate for ensuring the pressure of the low-pressure piping system under full RCS pressure (i.e., 15.4 MPa (2235 psig)), but no credit should be taken to consider these valves operable under such conditions without further justification.

Other Components in Low-Pressure Systems

For other components in the low-pressure systems, such as pumps, tanks, Hxs, flanges, and instrument lines, the staff finds that establishing an appropriate safety factor involves several complicating factors related to the individual component design. These factors include requirements for shop hydrotests, the method to determine the pressure class rating of the component, the specific material used for bolting and the bolt tension applied, or whether the component is qualified by test or analysis.

The remaining components in the low-pressure systems are designed to a design pressure of 0.4 times the normal operating RCS pressure (i.e., 6.16 MPa (894 psig)). The staff finds that the margins to burst for these remaining components are at least equivalent to those of the piping at its minimum wall thickness since these components typically have wall thicknesses greater than that of the pipe minimum wall thickness.

Conclusions

On the basis of the preceding considerations, the staff finds for the System 80+ low-pressure piping systems that interface with the RCS pressure boundary, that using a design pressure equal to 0.4 times the normal operating RCS pressure of 15.4 MPa (2235 psig) (i.e., 6.16 MPa (894 psig)), and using a minimum wall thickness of the low-pressure piping of no less than standard weight for piping of diameter 356 mm (14 in.) and less and no less than Schedule 40 for piping of diameter 406 mm (16 in.) and greater is an adequate basis for assuring that these systems can withstand full reactor pressure, and thus meets the Commission-approved requirements for designing against ISLOCA. The piping design is in accordance with the ASME Code, Section III, Subarticle NC/ND-3600. Furthermore, the staff will continue to require periodic surveillance and leak rate testing of the pressure isolation valves per TS requirements (Section 3.4.13) as a part of the ISI program.

As stated in SECY-90-016, for those low-pressure systems for which it is impractical to design to withstand full reactor pressure, as discussed above, the design provides (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room when isolation valve operators are de-energized, and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

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Using these minimum design guidelines, the staff reaches the following conclusions:

- (1) The likelihood of the low-pressure piping rupturing under full RCS pressure is low.
- (2) The likelihood of intolerable leakage is low under ISLOCA conditions, although some leakage might occur at flanges and valve bonnets.
- (3) Some piping components might undergo gross yielding and permanent deformation under ISLOCA conditions.

Because of the likelihood of piping system distortion, the staff finds that a visual inspection of the low-pressure piping system after it has been subjected to the full RCS pressure is necessary to identify those components that might require replacement, repair, or further detailed analysis before restart. The minimum design pressure for the low-pressure piping systems has been designated as a Tier 1 commitment for the System 80+.

On basis of this commitment, the staff concludes that there is reasonable assurance that the low-pressure piping systems interfacing with the RCPB are structurally capable of withstanding the consequences of an ISLOCA. The staff further concludes that the System 80+ design meets the Commission-approved staff recommendation for ISLOCAs, as an applicable regulation as described in Section 1.6 of this report.

3.9.3.2 Pump and Valve Operability Assurance

In its response to RAI Q210.60 and Q210.82, ABB-CE stated that it is in the process of revising CESSAR-DC Section 3.10 to conform to the guidelines of SRP Section 3.10 and RG 1.100 and that any necessary revisions to CESSAR-DC Section 3.9.3.2, to be consistent with the revised Section 3.10, would be supplied in a future submittal. By DSER Open Items 3.9.3.2-1 and 3.10-1, the staff noted that its review of CESSAR-DC Sections 3.9.3.2 and 3.10 was in progress. Subsequently, ABB-CE revised CESSAR-DC Sections 3.9.3.2 and 3.10 (Amendment R). Section 3.9.3.2 describes a comprehensive operability assurance program for pumps and valves. The program entails verifying (1) that safety-related pumps and valves are designed to be capable of performing all safety-related functions during and following design-basis events, (2) that analysis and/or tests are performed to demonstrate the operability of pumps and valves under the most severe postulated loadings including the SSE, (3) that each pump and valve is inspected to assure compliance with design specifications and drawings, (4) that each pump and valve is shop tested to verify "as-built" conditions, and (5) that

startup and IST is performed in accordance with ASME Code, Section XI, requirements to demonstrate that the active pumps and valves are operable throughout the life of the plant. Section 3.9.3.2 also identifies specific analyses and tests that must be done to qualify or to demonstrate the operability of ASME Code class pumps and valves, including pneumatically operated valves, motor-operated valves (MOVs), pressurizer safety valves, and check valves. CESSAR-DC Section 3.10 describes the methodologies, procedures, and criteria for the seismic and dynamic qualification testing of all safety and non-safety-related mechanical and electrical equipment and components, including pumps and valves.

The staff reports its evaluation of the design for ASME Code, Class 1, 2, and 3 components, including pumps and valves and their internals, the testing programs for pumps and valves, and the seismic and dynamic qualification program for all mechanical equipment and supports, in Sections 3.9.3.1, 3.9.6, and 3.10, respectively, of this report. On this basis, DSER Open Item 3.9.3.2-1 is resolved.

3.9.3.3 Design and Installation of Pressure-Relief Devices

CESSAR-DC Section 3.9.3.3 provides information relating to the design, installation, and testing criteria applicable to the mounting of pressure-relief devices used for the overpressure protection of ASME Code, Section III, Class 1, 2, and 3 components. The staff reviewed CESSAR-DC Section 3.9.3.3 in accordance with SRP Section 3.9.3. This review included an evaluation of the applicable loading combinations and stress criteria. The review considered the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens, and the transient fluid-induced loads applied to the piping downstream of a safety valve or relief valve in a closed discharge piping system. Except as noted in the discussion that follows, the information in CESSAR-DC Section 3.9.3.3 meets the applicable guidelines of SRP Section 3.9.3 and is, therefore, acceptable.

By RAI Q210.62, the staff noted that ABB-CE's use of criteria in the ANSI/ASME B31.1 Code, Appendix II, "Non-Mandatory Rules for the Design of Safety Valve Installations," as an alternative to performing a dynamic analysis was not acceptable. In response to the staff's request, ABB-CE committed to revising the CESSAR-DC to replace the reference to the criteria in the B31.1 Code with reference to an equivalent static analysis that follows the criteria of Appendix O of the ASME Code, Section III, as supplemented by the additional criteria of SRP Section 3.9.3, Item II.2. This was DSER Confirmatory Item 3.9.3.3-1. In Amendment K to CESSAR-DC,

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ABB-CE revised Section 3.9.3.3 to incorporate the preceding commitment. On this basis, DSER Confirmatory Item 3.9.3.3-1 is resolved.

On the basis of its review of CESSAR-DC Section 3.9.3.3, the staff finds that ABB-CE meets 10 CFR 50.55a and GDC 1, 2, and 3 with respect to the criteria to be used for the design and installation of ASME Code, Class 1, 2, and 3 overpressure-relief devices by ensuring that safety and relief valves and their installations will be designed to standards that are commensurate with their safety functions, and that they will accommodate the effects of discharge caused by normal operation as well as such postulated events as LOCAs and the dynamic effects resulting from the SSE. ABB-CE also meets GDC 14 and 15 with respect to ensuring that the RCPB design limits for normal operation, including anticipated operational occurrences, will not be exceeded. The criteria used by ABB-CE in the design and installation of ASME Code, Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure-relief devices provides a conservative basis for the design and installation of the devices for ensuring that the devices will withstand these loads without loss of structural integrity or impairment of the overpressure-protection function.

3.9.3.4 Component Supports

CESSAR-DC Section 3.9.3 describes ASME Code, Section III equipment supports as designed and constructed to either the ASME Code, Section III, Subsection NF or the AISC requirements. In accordance with SRP Table 3.2.2-1, the design and construction of supports of ASME Code, Section III, must be based on the ASME Code jurisdictional boundary limits defined by ASME Code, Section III, Subsection NF.

Moreover, CESSAR-DC Section 3.9.3.4 provides information on the specification for design of ASME Code, Section III, Class 1, 2, and 3 component supports. Supports design is in accordance with loads and loading combinations discussed in CESSAR-DC Section 3.9.3.1 and presented in CESSAR-DC Table 3.9-2. The early CESSAR-DC amendments committed ABB-CE to ASME Code, Section III, Subsection NF, allowable loads and stresses, but it did not specifically commit ABB-CE to construct (as defined in ASME Code, Section III, Article NCA-1110) supports to Subsection NF requirements. By DSER Open Item 3.9.3.4-1, ABB-CE was asked to commit to constructing component supports according to the Subsection NF requirements. In response, ABB-CE

revised CESSAR-DC Section 3.9.3.4 to specifically state that ASME Code Class 1, 2, and 3 component supports are designed and constructed in accordance with ASME Code, Section III, Subsection NF, requirements. This is acceptable and resolves DSER Open Item 3.9.3.4-1.

In addition, by RAI Q210.59, the staff asked ABB-CE to revise CESSAR-DC Section 3.9.3 to commit to the jurisdictional boundary between the component supports governed by ASME Code, Subsection NF, and the building structures as defined in the 1987 Addenda to the 1986 Edition of the ASME Code. The boundary in the 1987 Addenda is the same as in the 1989 ASME Code. Before the 1987 Addenda were issued, the staff had raised issues relating to the ASME Code defined jurisdictional boundary between Subsection NF components and building structures. The staff, on the basis of its review of the jurisdictional boundaries described in the 1989 Edition, finds that the boundaries are sufficiently defined to ensure a clear division between the pipe support and the structural steel and are thus acceptable. ABB-CE responded to RAI Q210.59 by committing to the jurisdictional boundary in accordance with ASME Code, Section III, Subsection NF of the edition specified by 10 CFR 50.55a. This response was not totally acceptable. In response to DSER Open Item 3.9.3.4-2, ABB-CE revised CESSAR-DC Section 3.9.3, in Amendment Q, to commit to the jurisdictional boundary between ASME Code, Subsection NF and building structures as defined in the 1989 Edition of the ASME Code that is specified by ABB-CE in CESSAR-DC Table 1.8-6. On this basis, DSER Open Item 3.9.3.4-2 is resolved.

By RAI Q210.65, the staff asked ABB-CE to provide information identified in SRP Section 3.9.3, Revision 1, Item II.7, relating to safety-related components that utilize snubbers. This information was to be in tabular form and to include the following data:

- identification of the systems and components in those systems that utilize snubbers
- the number of snubbers utilized in each system and components in that system
- the type(s) of snubber (hydraulic or mechanical) and the corresponding supplier
- whether or not the snubber was constructed to the rules of ASME Code, Section III, Subsection NF
- whether or not the snubber is used as a shock, vibration, or dual-purpose snubber

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- for snubbers identified as either dual-purpose or vibration arrester type, whether or not both snubber and components were evaluated for fatigue strength

ABB-CE responded to RAI Q210.65 with a listing of all safety-related components that utilize snubbers, including the requested details regarding plant arrangement, piping layout, and piping design. As presented to the staff at the meeting of November 26, 1991, ABB-CE stated that detailed piping system design and layout and plant arrangements (1) are not required for certification, (2) depend on plant-specific details not finalized at the certification stage, and (3) are subject to revision until specific piping, components, and other plant design feature details are finalized. By DSER COL Action Item 3.9.3.4-1, the staff requested that the COL applicant provide a listing of all safety-related components which utilized snubbers per SRP Section 3.9.3, Revision 1, Item II.7. In response, ABB-CE revised CESSAR-DC Section 3.9.3.4 to state that site-specific information will list all safety-related components which utilize snubbers in accordance with SRP Section 3.9.3. In addition, if energy-absorbing and/or non-linear piping restraints are used, a description of the methodology used to analyze and design the piping systems incorporating these elements will be provided. This is acceptable. This is included in COL Action Item 3.9-1.

By RAI Q210.66, the staff asked ABB-CE to make a commitment in CESSAR-DC Table 3.2-1 and CESSAR-DC Section 3.9.3.4 that concrete expansion anchor bolts which are used for pipe support base plates are designed to the applicable factors of safety and baseplate flexibility accountability requested in IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 2, November 8, 1979.

ABB-CE responded to RAI Q210.66 by stating that the requirements of ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures," are imposed and the issues identified in IE Bulletin 79-02, are addressed during detailed baseplate design to avoid problems experienced with concrete expansion anchor bolts.

ABB-CE stated that it would revise CESSAR-DC Section 3.9.3.4 to add the following:

Expansion anchors are designed in accordance with ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures." This assures that the design strength of concrete for a given expansion anchor or group of anchors is greater than the strength of the anchor steel, accounts for the effect of shear-tension interaction, and considers minimum edge distance and bolt spacing on expansion anchor capacity. In addition, base plate flexibility is

accounted for in the calculation of expansion anchor bolt loads.

This commitment was identified in the DSER because the staff at that time had not endorsed ACI-349, Appendix B. ACI-349, Appendix B, was being revised, in part to address certain unconservative approaches relating to expansion anchor capacities. By DSER Open Item 3.9.3.4-3, the staff noted that ABB-CE should readdress the issues presented in RAI Q210.66 to respond to the staff's concerns. The design of expansion anchors is primarily a structural issue. The adequacy of the design has been addressed by structural review and the staff review of this issue is discussed in Appendix 3B of this report. On this basis, DSER Open Item 3.9.3.4-3 is resolved as a mechanical issue.

Additionally, in response to RAI Q210.66 (and the related RAI Q210.1), ABB-CE revised CESSAR-DC Table 3.2-1 in Amendment N to add component supports under structures and to insert note 23, "Component supports are designed to the criteria described in Section 3.9.3.4," in the table.

On the basis of its review of CESSAR-DC Section 3.9.3.4, the staff finds that ABB-CE meets 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code, Class 1, 2, and 3 component supports.

3.9.4 Control Rod Drive Systems

In SRP Section 3.9.4, the staff defines the scope of and criteria for staff reviews of control rod drive systems (CRDSs). In SRP Section 3.9.4, the staff specifies that the CRDS consists of the control rods and related mechanical components that provide the means of mechanical movement and are required to be designed in accordance with GDC 26 and 27. Portions of the CRDS that are a part of the RCPB are required to be designed in accordance with GDC 1, 2, 14, and 29 and 10 CFR 50.55a. In SRP Section 3.9.4, the staff further specifies that the CRDS is considered to extend to where the coupling joins the reactivity control elements in the reactor pressure vessel (RPV), but that for electromagnetic systems, which is the case in the CESSAR-DC System 80+ design, the review of the CRDS should be limited to just the CRDM portion of the CRDS.

CESSAR-DC Section 3.9.4 provides information on the System 80+ CEDM design. The information is essentially identical to that in CESSAR Section 3.9.4 for the System 80 CEDM design. Detailed information relating to (1) the CEDM components and their operation, (2) applicable

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codes and standards, (3) design bases, and (4) the CEDM performance assurance program is provided in CESSAR-DC Sections 3.9.4.1 through 3.9.4.4, respectively.

The staff reviewed the description of the components and operation of the System 80+ CEDM in CESSAR-DC Section 3.9.4.1 and the information in CESSAR-DC Section 3.9.4.4 to evaluate the adequacy of the CEDM to perform its mechanical function properly during normal operating and postulated accident conditions. The staff noted in RAI Q210.70 that ABB-CE specifies a design life of 60 years for the System 80+ CEDM in CESSAR-DC Section 3.9.4.1 and a total cumulative control element assembly (CEA) travel of 30,480 m (100,000 ft) of operation without loss of function. The staff further noted, in DSER Open Item 3.9.4-1, that the CEA design travel is identical to that specified for the 40-year design life of the System 80 CEDM. ABB-CE responded in a letter dated December 23, 1992, that the design lifetime requirement as defined in CESSAR-DC Section 3.9.4.1 was a total cumulative CEA travel of 100,000 ft without loss of function and not for the 60-year plant life, and that the operational requirement for the System 80+ CEAs is expected to be less than the 100,000 ft of travel over the 60-year plant life. In CESSAR-DC Section 3.9.4.1, ABB-CE states that the CEDM is designed to operate without maintenance for a minimum of 1-1/2 years and without replacing components for a minimum of 3 years. This removal and maintenance schedule together with the replacement features of the CEDM motor assembly design provides acceptable justification for the 60-year design life. Additionally, portions of the CEDM that are part of the RCPB are designed for the 60-year plant design life as discussed in Section 3.9.1 of this report. This is acceptable and resolves DSER Open Item 3.9.4-1.

The staff also requested in RAI Q210.70 that ABB-CE submit the information regarding System 80 CEDM operating experience at Palo Verde referenced in CESSAR-DC Section 3.9.4.4.1.3, for demonstrating the wear resistance of the CEDM design. ABB-CE responded by deleting "measurable wear" from CESSAR-DC Section 3.9.4.4.1.3 on the basis that component wear was inferred from operational tests on CEDM motors only and did not apply to the CEDM design. This is acceptable to the staff because operational test results will provide sufficient data of wear resistance for CEDM motors.

Codes and Standards

The staff reviewed the codes and standards applicable to the CEDM as described in CESSAR-DC Section 3.9.4.2. Section 3.9.4.2 specifies that (1) pressure boundary components of the CEDM are "designed and fabricated" in

accordance with ASME Code, Section III, Class 1 requirements, (2) non-pressure boundary components are designed and verified on the basis of the previous prototype accelerated life testing described in CESSAR-DC Section 3.9.4.4, and (3) reed switch position transmitter (RSPT) assemblies are qualified in accordance with IEEE 323 and 344. This generally conforms to SRP Section 3.9.4 and SRP Section 3.10 and is acceptable.

By RAI Q210.72 and DSER Open Item 3.9.4-2, the staff asked ABB-CE to revise CESSAR-DC Section 3.9.4.2 in accordance with the criteria of SRP Section 3.9.4, Revision 2, Item II.2, including "construction" criteria. Consistent with RG 1.26, SRP Section 3.9.4 Item II.2 specifies that portions of the CEDM that are part of the primary pressure boundary and classified as Safety Class 1, Quality Group A, must be constructed in accordance with ASME Code, Section III, Class 1 requirements and explicitly specifies that "construction" must be as defined in NCA-1110 of Section III of the ASME Code. In response to DSER Open Item 3.9.4-2, ABB-CE revised Section 3.9.4.2 (Amendment N) to specify that the pressure boundary components are constructed in accordance with the requirements of ASME Code, Section III, Class 1 and the guidelines of SRP Section 3.9.4, Revision 2, Item II.2. This response is acceptable and resolves DSER Open Item 3.9.4-2.

For non-pressure-retaining components of the CEDM, design verification on the basis of testing is in accordance with SRP Section 3.9.4, Revision 2, Item I.2. Accelerated life tests performed on a pre-System 80 prototype CEDM are described in CESSAR-DC Section 3.9.4.4. The tests found identified a potential problem relating to excessive wear which resulted in changes intended to improve the design. More significantly, as stated in Section 3.9.4.4, the System 80+ CEDM is identical to those at Palo Verde that have been operating without malfunction. Although neither the accelerated life tests of the pre-System 80 CEDM nor the Palo Verde CEDM operating experience may be of sufficient duration to permit extrapolation to a 60-year CEDM design life, the CEDM maintenance and replacement design capabilities, the maintenance schedules specified in CESSAR-DC Section 3.9.4.1, the accelerated life tests, and the Palo Verde operating experience will serve as an adequate basis for the CEDM 60-year design life. However, the staff is aware of problems experienced with the CEDMs at Palo Verde over the past several years. Specifically, the design of the CEDM coil, which allowed for intermittent electrical grounding, resulted in multiple slip of CEAs. By DSER Open Item 3.9.4-3, ABB-CE was asked to address the CEDM problems to ensure that the design is not susceptible to the Palo Verde type of problems. In a letter dated December 23, 1993, ABB-CE stated that problems were encountered with lower latch

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coils in the Palo Verde CEDM coil stack assembly. The problem was corrected by the addition of insulating jackets to the coil leads and by reversing the polarity of the lower latch coil, which significantly reduced the intensity of adverse motion. These changes were instituted at Palo Verde and incorporated into the System 80+ CEDM design. This is acceptable and resolves DSER Open Item 3.9.4-3.

CESSAR-DC Section 3.9.4.2 specifies that the RSPTs are qualified in accordance with IEEE 323-1974 and 344-1975. In response to RAI Q210.61 and Q210.78, ABB-CE committed to revising these standards to IEEE 323-1983 and 344-1987 Editions as endorsed by current RG 1.100, Revision 2. However, the staff found that ABB-CE had been correct in its initial reference to IEEE 323-1974. Therefore, ABB-CE should not revise any references to IEEE 323 in the CESSAR-DC to specify the 1983 Edition; rather ABB-CE should revise these references to specify the 1974 Edition as endorsed by RG 1.89. This was DSER Open Item 3.9.4-4. Subsequently, the staff noted, in Amendment N to CESSAR-DC Section 3.9.4.2, that the RSPT assembly is designed to comply with IEEE 323-1974 and IEEE 344-1987. On this basis, DSER Open Item 3.9.4-4 is resolved. By RAI Q210.73, the staff also asked ABB-CE to clarify the seismic input motion(s) to be used for qualification in accordance with the IEEE 344 standard. ABB-CE responded that each item of equipment is to be (1) qualified on the basis of the envelope of the responses at all locations within the plant where the item may be located and for all generic soil types considered in the plant design basis, or (2) qualified separately for each soil type considered in the plant design basis. The response meets the guideline of SRP Section 3.10, II.1.a(3), and is acceptable.

ABB-CE's response to RAI Q210.73 also clarified that during testing, full-length RSPT assemblies are subjected to biaxial input motions and tests are performed in four RSPT orientations to account for asymmetries in the design.

On the basis of its review of CESSAR-DC Section 3.9.4, and codes and standards applicable to CRDM design, the staff concludes that the System 80+ CEDMs are expected to perform their mechanical function properly during the 60-year-plant-design life, and that positive means for inserting the rods will always be maintained to ensure appropriate margin for malfunction, such as stuck rods. The staff also concludes that the design codes and standards applicable to the CEDM are commensurate with the importance of the safety function of the CEDM and meets the requirements of GDC 1 and 10 CFR 50.55a(g).

Design Bases

CESSAR-DC Section 3.9.4.3 and Table 3.9-13 provide information on the CEDM design bases. The information includes (1) identification of the loads, loading combination, and stress limits for pressure boundary components, (2) testing for non-pressure boundary components, and (3) functional, including deformation, requirements.

By RAI Q210.74 and DSER Open Item 3.9.4-5, the staff noted that CESSAR-DC Section 3.9.4.3 and Table 3.9-13 of the early amendments did not contain LOCA loadings and asked ABB-CE to add these loadings. In response, ABB-CE revised CESSAR-DC Table 3.9-13 (Amendment T) to add the applicable LOCA loads due to pipe breaks not eliminated by LBB. This is acceptable and resolves DSER Open Item 3.9.4-5. In addition, by RAI Q210.74, the staff asked ABB-CE to state in CESSAR-DC Table 3.9-13 that dynamic loads should be combined by the SRSS method in accordance with NUREG-0484, Revision 1. In response to RAI Q210.74 and DSER Open Item 3.9.4-6, ABB-CE specified in CESSAR-DC Table 3.9-13 (Amendment Q) that dynamic loads include SSE, LOCA, and DBPBs and they are combined by the SRSS method in accordance with the guidelines of NUREG-0484. This is acceptable and resolves DSER Open Item 3.9.4-6.

On the basis of its review, the staff finds that the design bases for the CEDM are adequate to perform their safety function during anticipated operational occurrences and to withstand the effects of postulated accidents and such natural phenomena as earthquakes.

3.9.5 Reactor Pressure Vessel Internals

GDC 1, 2, 4, and 10 and 10 CFR 50.55a require that structures and components important to safety (such as reactor pressure vessel internals) be constructed and tested to quality standard commensurate with the importance of the safety functions to be performed, and designed with appropriate margins to withstand effects of anticipated normal plant occurrences, natural phenomena such as earthquakes and postulated accidents including LOCA.

In CESSAR-DC Section 3.9.5, ABB-CE identifies the components of the RPV core support and other ISs. ABB-CE describes the components of these structures and their arrangements in CESSAR-DC Section 3.9.5.1, and design loading conditions, design loading categories, and design bases in CESSAR-DC Sections 3.9.5.2 through 3.9.5.4, respectively.

The staff reviewed the description of the components of the reactor vessel internals and their arrangements (as

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given in CESSAR-DC Section 3.9.5.1) and noted the descriptions were in accordance with the guidance of SRP Section 3.9.5. The components of the reactor vessel ISs and their arrangements are substantially the same as in the System 80 design.

The staff reviewed the loading conditions considered in the design of the core support and ISs. By RAI Q210.75 and DSER Open Item 3.9.5-1, the staff noted that these loading conditions specified in CESSAR-DC Section 3.9.5.2 did not include the DBPB or main steam/feedwater pipe break (MS/feedwater pipe break), or LOCA loads described under the Level D service loadings in CESSAR-DC Section 3.9.5.3. Moreover, the staff noted that no information on these transients nor justification for limitation of consideration of transients to those not requiring forced shutdown were found in the early amendments of the CESSAR-DC. In response to DSER Open Item 3.9.5-1, ABB-CE revised CESSAR-DC Section 3.9.5.2 (Amendment N) for design loading conditions to include appropriate DBPB, secondary side breaks and LOCA loads as well as all anticipated transients. This is acceptable and resolves DSER Open Item 3.9.5-1.

The staff reviewed the loading combinations specified in CESSAR-DC Section 3.9.5.3 for the design of the core support and ISs. The loading combinations were specified for ASME Code Levels A, B, and D service conditions. The combinations included the effects of earthquakes and were sufficiently broad to include the effects of normal operations, maintenance, testing, and postulated LOCA, but emergency conditions were not identified for any Level C conditions. By DSER Open Item 3.9.5-2, the staff requested justification of the lack of Level C conditions for the applicable reactor internal components. In response, ABB-CE revised CESSAR-DC 3.9.5.3 (Amendment Q) to reference ASME Code, Level C, service conditions. This conforms to SRP Section 3.9.5 and resolves DSER Open Item 3.9.5-2.

The staff reviewed the design-basis stress limits specified in CESSAR-DC Table 3.9-14 for the core support (CS) and ISs. By RAI Q210.76, the staff noted that the original Table 3.9-14 of the CESSAR-DC did not indicate that the limits are applicable to ASME Code CS components and reactor vessel internals. In response to RAI Q210.76, ABB-CE committed to revise this table to state that (1) core support structures should be constructed to ASME Code, Section III, Subsection nitroglycerin/noble gas (NG)-1100 requirements, and (2) reactor internals, other than core support structures, should meet the guidelines of NG-3000 and be constructed so as not to impair the integrity of the core support structures. Also, in response to RAI Q210.77, ABB-CE committed to further revise this table to state that for Level D service conditions, stresses

will be obtained from an SRSS combination of LOCA and SSE loadings plus normal dynamic and static operating loads in accordance with NUREG-0484, Revision 1. For other than Level D loading conditions, the maximum stress intensity will also be derived in accordance with NUREG-0484, Revision 1 or a more conservative summation of stress intensities. By DSER Open Item 3.9.5-3, the staff noted that CESSAR-DC Table 3.9-14 should be revised to be applicable to all dynamic loads. In response to DSER Open Item 3.9.5-3, ABB-CE revised CESSAR-DC Table 3.9-14, Amendment T, to incorporate these revisions. On this basis, DSER Open Item 3.9.5-3 is resolved.

The staff reviewed the deformation limits and their bases specified in CESSAR-DC Section 3.9.5.4 for proper functioning of the reactor ISs. These limits and their bases are the same as those previously specified for the System 80 plant design and, hence, acceptable.

However, in the design of critical reactor vessel internals, in the early amendments of CESSAR-DC Section 3.9.5.4, ABB-CE stated that for components subject to fatigue, the stress analysis will be performed using the design fatigue curve of I-9-2 of Section III of the ASME Code. The staff raised concerns relating to possible detrimental environmental effects not presently reflected in current ASME Code design fatigue curves (see RAI Q210.46 and RAI Q210.52). Subsequently, in CESSAR-DC Section 3.9.1.1 (Amendment R), ABB-CE noted that observations of significant environmental degradation in LWR environments are primarily related to high strain ranges, slow strain rates, high oxygen contents of primary water, high sulfur contents of carbon and low alloy steels, and low-flow-rate conditions. The CESSAR-DC further stated that, since System 80+ components are not exposed to high oxygen content environments at elevated temperatures, and no carbon or low alloy steel is directly exposed to the primary coolant, significant environmental degradation would not likely occur to the System 80+ components which include reactor vessel internals. This clarification is acceptable and resolves DSER Open Item 3.9.5-4.

On the basis of this review, the staff concludes that the design of reactor internals is acceptable and meets the requirements of GDC 1, 2, 4, and 10 and 10 CFR 50.55a as discussed below.

- ABB-CE meets the requirements of GDC 1 and 10 CFR 50.55a with respect to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are in conformance with the requirements of Subsection NG of the ASME Code, Section III.

- ABB-CE meets the requirements of GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquake and the effects of NOP, maintenance, testing, and postulated LOCAs with sufficient margin to ensure that capability to perform their safety functions is maintained and the specified acceptable fuel design limits are not exceeded.
- The specified design transients, design, service loadings, and combination of loadings as applied to the design of the reactor internals structures and components offer reasonable assurance that in the event of an earthquake or a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components will not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events that have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

3.9.6 Testing of Pumps and Valves

SRP Section 3.9.6 provides guidance for review of IST of certain safety-related pumps and valves typically designated as ASME Code Class 1, 2, or 3. Section 3.9.3 of this report discusses the design of safety-related pumps and valves in ABB-CE System 80+. The load combinations and stress limits used in the design of pumps and valves ensure that the integrity of the component pressure boundary will be maintained. In addition, ABB-CE will periodically test the performance and will measure performance parameters of safety-related pumps and valves. The tests and measurements will be performed in accordance with Section XI of the ASME Code as required by 10 CFR 50.55a(f). Periodic measurements of various parameters will be compared to baseline measurements to detect long-term degradation of the pump or valve performance.

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," January 12, 1990, the staff concluded that the requirements of Section XI of the ASME Code provide information on the operational readiness of the components, but, in general, do not necessarily verify the capability of the components to perform their intended safety functions. The staff concluded that the ASME Code does not assure the necessary level of component operability that is desired for the evolutionary LWR designs.

Accordingly, in SECY-90-016, as supplemented by the staff's April 27, 1990, response to comments by the ACRS, the staff recommended criteria to the Commission to be used to supplement Section XI of the ASME Code. In addition, the staff agreed with the recommendation of the ACRS that the guidelines of Generic Letter (GL) 89-10 "Safety-Related Motor Operated Valve Testing and Surveillance," be applied to evolutionary LWR designs. The ACRS also recommended that the staff resolve the check valve testing and surveillance issue and that consideration be given to industry-proposed alternative ways of meeting IST and surveillance requirements. In its SRM of June 26, 1990, on SECY-90-016, the Commission approved the staff's recommendations. The staff's proposed applicable regulation, as described in Section 1.6 of this report, for IST, is as follows.

- (1) Piping design must incorporate provisions for full flow testing at maximum design flow of pumps and check valves;
- (2) Check valve testing must incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics;
- (3) Provisions must be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation which cannot be detected through the use of advanced non-intrusive techniques; and
- (4) Provisions must be incorporated provisions to test MOVs under design-basis differential pressure.

The staff's evaluation of ABB-CE's responses to ASME Code Section XI issues, including the above applicable regulations, is discussed in the sections below.

By DSER Open Item 3.9.6-1, the staff asked ABB-CE to submit an IST plan for pumps and valves and to clarify the scope of the IST. In CESSAR-DC Table 3.9-15, Amendment R, ABB-CE submitted its IST plan for safety-related pumps and valves. In Section 3.9.6 of the CESSAR-DC, Amendment R, ABB-CE clarified that its IST plan as submitted in CESSAR-DC Table 3.9-15 includes those safety-related pumps and valves that are necessary to ensure

- the integrity of the RCPB
- the capability to achieve safe shutdown of the reactor and keep it in a safe-shutdown condition

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- the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures in excess of 10 CFR Part 100 guidelines

The staff's assessment of the IST plan is discussed in Section 3.9.6.3 of this report. On this basis, DSER Open Item 3.9.6-1 is resolved.

In the DSER, the staff noted COL Action Item 3.9.6-1 relating to plant-specific IST program. The staff noted that the development of a complete plant-specific IST program shall be the responsibility of the COL applicant. The comprehensive pump and valve IST program will provide additional information beyond that contained in the IST plan. The IST program will include: the tests performed on each pump and valve and the Code requirement met by each test; test parameters and frequency of the tests; the normal, safety, and fail-safe position of each valve; component type for each pump and valve; and P&ID coordinates for each pump and valve. In addition, any requests for relief shall be submitted by the COL applicant and will be reviewed by the NRC staff, on the basis of the Code edition referenced in 10 CFR 50.55a(f), the System 80+ design, and the IST methods available at the time of the COL application. In Section 3.9.6 of the CESSAR-DC, Amendment R, ABB-CE stated that the COL applicant will submit a detailed pump and valve IST program. This is acceptable. This is included in COL Action Item 3.9-1.

By DSER Open Item 3.9.6-2, the staff asked ABB-CE to include provisions in its piping system design to accommodate the applicable ASME Code IST requirements. In Section 3.9.6 of the CESSAR-DC, Amendment R, ABB-CE stated that the IST plan as provided in Table 3.9-15 is developed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1,6, and 10. ASME/ANSI Part 6 of Operations and Maintenance (OM), "In-service testing of Pumps," and Part 10, "In-service testing of Valves," were referenced in Section XI, ASME Code, 1989 Edition. The 1988 Addenda and the 1989 Edition of Section XI have been incorporated by reference into 10 CFR 50.55a and are acceptable for the evolutionary LWR IST provided the analysis of leakage rates and corrective action requirements of OM Part 10, paragraph 4.2.2.3 are applied to CIV testing. The staff's evaluation of CIV leak testing is discussed in Section 3.9.6.2.4 of this report. On this basis, DSER Open Item 3.9.6-2 is resolved.

3.9.6.1 Testing of Safety-Related Pumps

GDC 1 requires that components important to safety be designed and tested to standards commensurate with the

importance of the safety functions to be performed. In response to the staff's concern regarding the adequacy of design and qualification for safety-related pumps for assuring their capability to perform design-basis functions, ABB-CE stated in Section 3.9.6.1 of the CESSAR-DC, Amendment R, that for each safety-related pump, the design basis and required operating conditions (including tests) under which the pump will be required to function will be established. These design conditions include flow rate and corresponding pump head for each system mode of pump operation and the required operating time for each mode, acceptable bearing vibration levels, seismic/dynamic loads, fluid temperature, ambient temperature, and pump motor minimum voltage. ABB-CE further stated that the COL applicant will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. For each size, type, and model of safety-related pump installed, the COL applicant will perform testing encompassing design conditions that demonstrate acceptable flow rate and corresponding head, bearing vibration levels, and pump internals wear rates for the operating time specified for each system mode of pump operation. From these tests the COL applicant will also develop baseline hydraulic and vibration data for evaluating the acceptability of the pump after installation. The staff finds these actions provide reasonable assurance for the adequacy of the design and qualification for safety-related pumps and are, therefore, acceptable.

In response to DSER Confirmatory Item 3.9.6.1-1 regarding the adequacy of minimum-flow systems for safety-related pumps, ABB-CE stated in Section 3.9.6.1 of the CESSAR-DC, Amendment R, that safety-related pumps and piping configurations can accommodate IST at a flow rate at least as large as the maximum design flow for the pump. This approach is responsive to the guidelines on this issue contained in SECY 90-016 and meets Item 1 of the staff's recommendations, as an applicable regulation, on pump and valve IST issues as described in Sections 1.6 and 3.9.6 of this report. This is acceptable. In addition, ABB-CE also stated that the COL applicant will ensure that the pump specified for each application is not susceptible to inadequate minimum flow rate and inadequate thrust bearing capacity. With respect to minimum pump flow operation, the sizing of each minimum recirculation flow path is evaluated to ensure that its use under all analyzed conditions will not degrade the pump. The flow rate through minimum recirculation flow paths can also be periodically measured to verify that it is in accordance with the design specification. In the same CESSAR-DC section, ABB-CE also stated that it would provide the safety-related pumps with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. The staff finds these commitments acceptable because they

meet SRP Section 3.9.6 testability requirements to permit appropriate periodic functional testings for safety-related pumps. On this basis, DSER Confirmatory Item 3.9.6.1-1 is resolved.

In Section 3.9.6.1 of the DSER, the staff identified DSER Open Item 3.9.6.1-1 relating to disassembly and inspection of safety-related pumps and valves. In Section 3.9.6.1 of CESSAR-DC Amendment R, ABB-CE stated that the detailed IST program will establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related pumps, including the basis for the frequency and the extent of each disassembly. Factors to be considered in the disassembly frequency and extent of disassembly include historical performance of the pump, pump service condition, and results of non-intrusive pump testing. ABB-CE further stated that the program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. This clarification of the actions by the COL applicant is responsive to the guidelines on this issue in SECY 90-016 and meets Item 3 of the staff's recommendations, as an applicable regulation, on the pump and valve IST issues as discussed in Sections 1.6 and 3.9.6 of this report. This is acceptable. Furthermore, the staff notes that these actions are also applicable to all safety-related valves as described in Sections 3.9.6.2.1.3, 3.9.6.2.2.3, and 3.9.6.2.3.3 of the CESSAR-DC, Amendment R. On this basis, DSER Open Item 3.9.6.1-1 is resolved.

3.9.6.2 Testing of Safety-Related Valves

3.9.6.2.1 Motor-Operated Valves

3.9.6.2.1.1 Design and Qualification Requirements for Motor-Operated Valves

In the DSER, the staff identified DSER Open Item 3.9.6.2-4 regarding the adequacy of design, qualification, and testing for safety-related MOVs for assuring their capability to perform design-basis functions. In Section 3.9.6.2.1 of the CESSAR-DC, Amendment R, ABB-CE stated that for each MOV performing an active safety-related function, the design basis and required operating conditions (including testing) under which the MOV will be required to perform are established for the development and implementation of the design, qualification and preoperational testing. For the design and qualification of MOVs, the CESSAR-DC states the following commitments. The COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model of MOV, the COL applicant will determine the torque and thrust requirements to operate the MOV and will ensure the adequacy of the torque and

thrust that the motor operator can deliver under design conditions. The COL applicant will also test each size, type, and model of MOV under a range of differential pressure and flow conditions up to the design conditions. These design conditions include fluid flow, differential pressure (including pipe break), system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and maximum stroke time requirements. This testing of each size, type, and model of MOV shall include test data from the manufacturer, field test data for plant-specific dedication, empirical data supported by test, or test of similar valves (such as prototype) that support qualification of the required valve where similarity must be justified by technical data. From this testing, the COL applicant will demonstrate that the results of testing under in situ or installed conditions can be used to ensure the capability of the MOV to operate under design conditions. The COL applicant will also ensure that the structural capability limits of the individual parts of the MOV will not be exceeded under design conditions. Furthermore, the COL applicant will ensure that the valve specified for each application is not susceptible to pressure locking and thermal binding. The staff finds these commitments meet the staff's guidelines prepared for the design and qualification for the safety-related MOVs of the evolutionary LWRs and will provide reasonable assurance for the adequacy of the design and qualification for safety-related MOVs and are, therefore, acceptable. The staff's assessment of the MOV preoperational testing and IST are contained in Sections 3.9.6.2.1.2 and 3.9.6.2.1.3 of this report.

3.9.6.2.1.2 Preoperational Testing of Safety-Related Motor-Operated Valves

For the preoperational testing of MOVs, the CESSAR-DC states commitments as follows. The COL applicant will test each MOV in the open and closed directions under static and maximum achievable conditions up to design-basis conditions using diagnostic equipment that measures torque and thrust, and motor parameters. The COL applicant will test the MOV under various differential pressure and flow up to maximum achievable conditions and perform a sufficient number of tests to determine the torque and thrust requirements at design conditions. The specific testing parameters and acceptance criteria for demonstrating that the adequacy of the MOV functional performance have been met are in CESSAR-DC Section 3.9.6.2.1.2. The staff finds these commitments to be acceptable because these meet the staff's guideline for the preoperational testing of safety-related MOVs. It should be noted that these commitments, as discussed in Sections 3.9.6.2.1.1 and 3.9.6.2.1.2 of this report, also relate to the MOV ITAAC item for demonstrating the adequacy of the MOV capability for the design-basis conditions and any change to these commitments would involve an

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unreviewed safety question and, therefore, requires NRC review and approval before implementation. Any requested change to these commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance. The periodic verification of the MOV design-basis capability is addressed in Section 3.9.6.2.1.3 of this report.

3.9.6.2.1.3 IST of Safety-Related Motor-Operated Valves

In response to the staff's concern regarding the periodic testing for MOVs, ABB-CE stated in Section 3.9.6.2.1.3 of the CESSAR-DC, Amendment R, that periodic testing per GL 89-10 Paragraphs D and J will be conducted under adequate differential pressure and flow conditions which allow a justifiable demonstration of continuing MOV capability for design-basis conditions. ABB-CE further stated that the detailed IST program will include the optimal frequency of this periodic verification. The frequency and test conditions will be sufficient to demonstrate continuing design-basis and required operating capability. This is responsive to the guideline on MOV testing contained in SECY-90-016 and meets Item 4 of the staff's recommendations, as an applicable regulation, on the pump and valve IST issues as described in Sections 1.6 and 3.9.6 of this report and is acceptable. ABB-CE also stated that the code provides criteria limits for the test parameters noted in CESSAR-DC Table 3.9-15 for code IST.

In Section 3.9.6.2.1.3 of the CESSAR-DC, Amendment R, ABB-CE stated that IST of MOVs will rely on diagnostic techniques that are consistent with the state of the art and that will permit an assessment of the performance of the valve under actual loading conditions. This is responsive to the staff's guidelines on this issue and is, therefore, acceptable. The staff's evaluations are described in Sections 3.9.6.2.1.1, 3.9.6.2.1.2, and 3.9.6.2.1.3 of this report. On this basis, DSER Open Item 3.9.6.2-4 is resolved.

3.9.6.2.2 Power-Operated Valves

3.9.6.2.2.1 Design and Qualification Requirements for Power-Operated Valves

In the DSER, the staff identified Open Item 3.9.6.2-5 regarding the adequacy of design, qualification, and testing for safety-related power-operated valves (POVs) other than MOVs for assuring their capability to perform design-basis functions. In Section 3.9.6.2.2 of the CESSAR-DC, Amendment R, ABB-CE stated that for each POV with active safety-related functions, the design-basis and required operating conditions (including testing) under

which the POV will be required to perform will be established. In Section 3.9.6.2.2.1 of the CESSAR-DC, Amendment R, ABB-CE further stated that the COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model of safety-related POV, the COL applicant will determine the force requirements to operate the POV and will ensure the adequacy of the force that the operator can deliver under design conditions. The COL applicant will also test each size, type, and model under a range of differential pressure and flow conditions up to the design conditions. This testing of each size, type, and model of POV shall include test data from the manufacturer, field test data for plant-specific dedication, empirical data supported by test, or test of similar valves (such as prototype) that support qualification of the required valve where similarity must be justified by technical data. From this testing, the COL applicant will demonstrate that the results of testing under in-situ conditions can be used to ensure the capability of the POV to operate under design conditions. The COL applicant will also ensure that the structural capability limits of the individual parts of the POV will not be exceeded under design conditions. The COL applicant will ensure that packing adjustment limits are specified for the valve for each application such that it is not susceptible to stem binding. The staff finds these actions meet the staff's guidelines prepared for the design and qualification for safety-related POVs of the evolutionary LWRs and will provide reasonable assurance for the adequacy of the design and qualification for safety-related POVs and are, therefore, acceptable. The staff's evaluation of the POV preoperational testing and IST is in Sections 3.9.6.2.2.2 and 3.9.6.2.2.3 of this report.

3.9.6.2.2.2 Preoperational Testing of Safety-Related POV

In Section 3.9.6.2.2.2 of the CESSAR-DC, Amendment R, ABB-CE stated that during preoperational testing, the COL applicant will test each POV in the open and close directions under static and maximum achievable conditions using diagnostic equipment that measures or provides information to determine total friction, stroke time, seat load, spring rate, and travel under normal pneumatic or hydraulic pressure and minimum pneumatic or hydraulic pressure. The COL applicant will test the POV under various differential pressure and flow, up to maximum achievable conditions, including design-basis conditions. The COL applicant will perform a sufficient number of tests to determine the force requirements at design conditions. The specific testing parameters and acceptance criteria for demonstrating that the adequacy of the POV functional performance have been met are provided in CESSAR-DC Section 3.9.6.2.2.2. The staff finds these

actions acceptable because they meet the staff's guideline for the preoperational testing of safety-related POVs of the evolutionary LWRs.

3.9.6.2.2.3 IST of Safety-Related POV

In Section 3.9.6.2.2.3 of the CESSAR-DC, Amendment R, ABB-CE stated that all safety-related piping systems incorporate provisions for testing to demonstrate the operability of the POVs under design-basis conditions. IST will incorporate the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of POVs. These actions are responsive to the staff's guidelines prepared for the IST of safety-related POVs of the evolutionary LWRs on these issues and are acceptable.

On the basis of the staff's evaluations as described in Sections 3.9.6.2.2.1, 3.9.6.2.2.2, and 3.9.6.2.2.3 of this report, DSER Open Item 3.9.6.2-5 is resolved.

3.9.6.2.3 Check Valves

3.9.6.2.3.1 Design and Qualification Requirements for Check Valves

In response to the staff's concern regarding the adequacy of design and qualification for safety-related check valves, ABB-CE stated in Section 3.9.6.2.3 of the CESSAR-DC, Amendment R, that for each check valve performing active safety-related function, the design-basis and required operating conditions (including testing) under which the check valve will be required to perform will be established. In Section 3.9.6.2.3.1 of the CESSAR-DC, Amendment R, ABB-CE further stated that the COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model of check valve, the COL applicant will ensure the design adequacy of each check valve under design conditions including severe transient loadings expected during the life of the valve such as waterhammer or pipe break. This testing of each size, type, and model of check valve shall include test data from the manufacturer, field test data for plant-specific dedication, empirical data supported by test, or test (such as prototype) of similar valves that support qualification of the required valve where similarity must be justified by technical data. The COL applicant will ensure that the maximum loading on the check valve under design basis and the required operating conditions is within the structural capability limits for the individual parts of the check valve. In addition, the COL applicant will ensure proper check valve applications in the piping system design. Specific considerations will include selection of valve size and type based on the system flow conditions,

installed location of valve with respect to source of turbulence, and correct orientation of valve in the system (e.g., vertical versus horizontal) as recommended or required by the valve manufacturer. The qualification acceptance criteria will include baseline data developed during qualification testing and will be used for verifying the acceptability of the check valve after installation. The staff finds these actions meet the staff's guidelines prepared for the design and qualification for safety-related check valves of the evolutionary LWRs and will provide reasonable assurance for the adequacy of the design and qualification for safety-related check valves and are therefore acceptable.

3.9.6.2.3.2 Preoperational Testing of Check Valves

In Section 3.9.6.2.3.2 of the CESSAR-DC, Amendment R, ABB-CE stated that during the preoperational testing, the COL applicant will test each safety-related check valve in the open and/or close direction, as required by the safety function, under all normal operating system conditions. The testing requirements and acceptance criteria including leak-tightness, disk stability, and correct valve sizing are in CESSAR-DC Section 3.9.6.2.3.2. The staff finds these actions acceptable because these meet the staff's guideline prepared for the preoperational testing of safety-related check valves of the evolutionary LWRs.

3.9.6.2.3.3 IST of Safety-Related Check Valves

In response to DSER Open Item 3.9.6.2-1 regarding the full-flow testing of check valves, ABB-CE stated that System 80+ safety-related piping systems design incorporates provisions for testing to demonstrate the operability of the check valves under design-basis conditions. In response to DSER Open Item 3.9.6.2-3, ABB-CE stated that advanced nonintrusive techniques will be used in the implementation of IST program to periodically assess degradation and the performance characteristics of check valves. These are responsive to the guidelines on these issues contained in SECY-90-016 and meet Items 1 and 2 of the staff's recommendations, as an applicable regulation, on the pump and valve IST issues as described in Sections 1.6 and 3.9.6 of this report and are, therefore, acceptable. On the basis of this evaluation, DSER Open Item 3.9.6.2-1 and 3.9.6.2-2 are resolved.

In the DSER, the staff identified DSER Open Item 3.9.6.2-2 relating to reverse-flow testing for check valve. The staff noted that the reverse-flow testing includes the leak rate testing for safety-related check valves designated Category A/C as well as the quarterly reverse-flow closure verification for safety-related check valves performing a safety function in the closed position whether categorized as C or A/C. Verification that a Category C

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valve is in the closed position can be done by visual observation, by an electrical signal initiated by a position-indicating device, by observation of appropriate pressure indication in the system by leak testing, or by other positive means. Furthermore, in accordance with OMa-1988, Part 10, as referenced by the 1989 Edition of ASME Code, Section XI, valve closure verification that is impractical to perform during power operation may be deferred to cold shutdown. If it is also impractical to perform at cold shutdown, it may be deferred to refueling. In CESSAR-DC Amendment R, ABB-CE revised its IST plan to reflect this staff position. On this basis, DSER Open Item 3.9.6.2-2 is resolved.

3.9.6.2.4 Isolation Valve Leak Tests

In Section 3.9.6.2.4 of CESSAR-DC Amendment R, ABB-CE stated that the leak-tight integrity of each valve relied upon to provide a leak-tight function will be verified. These valves include (1) pressure isolation valves that provide isolation of pressure differential from one part of a system to another or between systems, (2) temperature isolation valves whose leakage may cause unacceptable thermal stress fatigue or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps, and (3) CIVs that provide isolation capability for the piping system penetrating containment. The staff's evaluation of CESSAR-DC Section 3.9.6.2.4 follows.

In the DSER, the staff identified DSER Open Item 3.9.6.2-6 regarding the leak-rate testing requirement for CIVs. In Section 3.9.6.2.4 of the CESSAR-DC, Amendment R, ABB-CE stated that CIVs with their required testing are presented in CESSAR-DC Table 6.2.4-1. ABB-CE further stated that those valves for which the Appendix J, Type C test is specified in Table 6.2.4-1 will also be tested in accordance with OM-10, Subsections 4.2.2.3(e) and 4.2.2.3(f). The staff finds these commitments acceptable because these meet the staff's position as described in GL 89-04 for the leak-rate testing of CIVs. On this basis, DSER Open Item 3.9.6.2-6 is resolved. However, it should be noted that the compliance with GDC 54 for Table 6.2.4-1 is being reviewed under SRP Section 6.2.4 and is addressed in Section 6.2.4 of this report.

Several safety systems connected to the RCPB have design pressures below the rated RCS pressure. Also, some systems that are rated at full-reactor pressure on the discharge side of pumps have pump suction below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves will be placed in series to form the interface between the high-pressure RCS and the low-pressure system. In the DSER, the staff identified the

scope (valve list) and the surveillance requirements of pressure isolation valves as Open Item 3.9.6.2-7. In CESSAR-DC Table 3.9-16 Amendment T, ABB-CE listed RCS pressure isolation valves. In CESSAR-DC Section 3.9.6.2.4 ABB-CE also stated that those pressure isolation valves will be leak-rate tested in accordance with CESSAR-DC Table 3.9-15 and the surveillance requirements specified in TS 3.4.14.1. The staff finds the list of RCS pressure isolation valves acceptable. Furthermore, the periodic leak rate testing requirements for ABB-CE System 80+ RCS pressure isolation valves listed in the CESSAR-DC, Amendment U, are the same as the relevant surveillance requirements of the new ABB-CE standard TS and are acceptable. The staff also notes that, as a part of the resolution of the ISLOCA issue, ABB-CE commits to requiring that low-pressure piping systems which interface with the RCPB be designed to withstand forty percent of the full RCS pressure. The staff's evaluation of the ISLOCA design for ABB-CE System 80+ piping systems is acceptable as discussed in Section 3.9.3.1.1 of this report. Therefore, DSER Open Item 3.9.6.2-7 is resolved.

3.9.6.3 Review of CESSAR-DC Table 3.9-15, IST Plan

As discussed previously in Section 3.9.6 of this report, ABB-CE made certain commitments regarding the testability of safety-related pumps and valves in the System 80+ design. In CESSAR-DC Section 3.9.6, ABB-CE states that the IST plan as provided in CESSAR-DC Table 3.9-15 is developed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1, 6, and 10. CESSAR-DC Table 3.9-15 lists the individual pumps and valves and provides information for each component including valve type, code category, safety class, test parameters, test frequency, and test configuration number. Table 3.9-15 also provides explanatory notes and justifications for any ASME Code-defined testing exceptions. The staff, with the assistance of Brookhaven National Laboratory, reviewed and evaluated the System 80+ IST plan as presented in Table 3.9-15. Details of the evaluation of the IST plan are in Appendix 3C of this report. On the basis of that evaluation, the staff determined that the System 80+ pump and valve IST plan provided reasonable assurance that ABB-CE can fulfill its commitment as described in CESSAR-DC Section 3.9.6 regarding the ability to test pumps and valves, and is acceptable. On this basis, DSER Open Item 3.9.6-1 is resolved.

3.9.6.4 Conclusion

On the basis of preceding evaluations, described above, the staff concludes that ABB-CE's approach to pump and valve testing is acceptable and meets the requirements of GDC 37, 40, 43, 46, and 54 and 10 CFR 50.55a(f). This

conclusion is based on the staff's determination that ABB-CE's approaches regarding the testability of safety-related pumps and valves in the System 80+ design will ensure that safety-related pumps and valves are in a state of operational readiness to perform necessary safety functions throughout the life of the plant. ABB-CE's pump and valve testing program includes baseline pre-service testing, periodic IST, functional testing of the components in the operating state, and visual inspection for leaks and other signs of distress. The staff further concludes that the System 80+ plant meets Commission-approval staff position of SECY-90-016 and its applicable regulation for IST of pumps and valves as described in Sections 1.6 and 3.9.6 of this report. This is acceptable.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

CESSAR-DC Sections 3.9.2.2 and 3.10 provide information on the seismic and dynamic qualification of safety-related mechanical and electrical equipment. CESSAR-DC Section 3.9.3.2 also contains information relative to pump and valve operability assurance programs. This information includes

- rationale used to determine if tests, analyses, or combinations of both will be performed
- criteria used to define the seismic and other relevant dynamic load input motions
- the proposed demonstration of the adequacy of the qualification program

ABB-CE will use the seismic qualification methodology described in CESSAR-DC Section 3.10 for both mechanical and electrical equipment. This methodology conforms to IEEE 323 and also meets the criteria in IEEE 344-1987 and RG 1.100, Revision 2. It should be noted that Section 9 of IEEE 344-1987 recognized the use of "experience data" as a method for seismic qualification of equipment. As used in IEEE 344, experience data include both seismic experience and previous qualifications. In accordance with RG 1.100, Revision 2, the method of qualification using experience data will be reviewed by the staff on a case-by-case basis. CESSAR-DC Subsection 3.10.1.2, "Selection of Qualification Method," states that qualification of equipment by experience shall be used only on a case-by-case basis. It further states that the COL applicant shall submit documentation describing the class of equipment and the methodology to be used. This is included in COL Action Item 3.10-1.

The staff concludes that this commitment is consistent with the applicable portion of RG 1.100, Revision 2, and is acceptable. The staff also concludes that the criteria described in CESSAR-DC Section 3.10 are consistent with the guidelines in SRP Section 3.10 and applicable portions of SECY-93-087, and are acceptable. To provide a more detailed basis for the staff's conclusion, listed below are applicable guidelines from SRP Section 3.10 that CESSAR-DC Sections 3.10 adopts for the System 80+ design.

- (1) Tests and analyses are required to confirm the operability of all mechanical and electrical equipment during and after an earthquake of magnitude up to and including the SSE, and for all static and dynamic loads from normal, transient, and accident conditions. Before SSE qualification, it should be demonstrated that the equipment can withstand excitation of five 1/2 SSE events without loss of structural integrity. Analyses alone, without testing, are acceptable as a basis for qualification only if the necessary functional operability of the equipment is assured by its structural integrity alone. When complete testing is impractical, a combination of tests and analyses is acceptable.

Equipment that has been previously qualified by means of tests and analyses equivalent to those described here are acceptable provided that proper documentation of such tests and analyses exists.

- (2) Equipment should be tested in the operational condition. Operability should be verified during and/or after the testing, as applicable to the equipment being tested. Loadings simulating those of plant NOP, such as thermal and flow-induced loading, if any, should be concurrently superimposed upon the seismic and other pertinent dynamic loading to the extent practicable. Particular attention should be paid, in operability qualification of mechanical equipment subjected to flow-induced loading, to incorporate degraded flow conditions such as those that might be encountered from the presence of debris, impurities, and contaminants in the fluid system. An example of this may be the operability of the containment sump pump recirculating water full of debris.
- (3) The characteristics of the required seismic and dynamic input motions should be specified by response spectrum or time history methods. These characteristics, derived from the structures, or systems, seismic and dynamic analyses, should be representative of the input motions at the equipment mounting locations.

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- (4) For seismic and dynamic loads, the actual test input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated (i.e., the test response spectrum should closely resemble and envelope the required response spectrum over the critical frequency range).
- (5) Since the seismic and dynamic load excitations generally have a broad frequency content, multi-frequency vibration input motion should be used. However, single frequency input motion, such as sine beats, is acceptable provided that the characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects), or the anticipated response of the equipment is adequately represented by one mode, or in the case of structural integrity assurance, the input has sufficient intensity and duration to produce sufficiently high levels of stress for such assurance. Components that have been previously tested to IEEE 344-1971 should be re-evaluated to justify the appropriateness of the input motion used, and requalified if necessary.
- (6) For the seismic and dynamic portion of the loads, the test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously, unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

Components that have been previously tested to IEEE 344-1971 should be requalified using biaxial test input motions, unless justification for using a single axis test input motion is provided.

- (7) Dynamic coupling between the equipment and related systems, if any, such as connected piping and other mechanical components, should be considered. The fixture design should simulate the actual service mounting and should not cause any extraneous dynamic coupling to the test item.

- (8) For pumps and valves, the loads imposed by the attached piping should be properly taken into account. In order to ensure operability under combined loadings, the stresses resulting from the applied test loads should envelope the specified service stress limit for which the component's operability is intended.
- (9) Selection of damping values for equipment to be qualified should be made in accordance with RG 1.61 and IEEE 344-1987. Higher damping values may be used if justified by documented test data with proper identification of the source and mechanism. Sections 3.7, 3.9.2.2, 3.9.3.2, and 3.10.2 contain criteria that address this issue.
- (10) The test program is based upon selectively testing a representative number of components according to type, load level, size, or some other appropriate classification, on a prototype basis.

CESSAR-DC Section 3.10 contains qualification methodology only; it has no plant-specific information. Therefore, each COL applicant using this methodology must ensure that specific environmental parameters along with seismic and dynamic input response spectra are properly defined and enveloped in the methodology for its specific plant and implemented in its equipment qualification program. In CESSAR-DC Sections 3.10.4 and 3.10.5, ABB-CE commits that the COL applicant shall maintain equipment qualification records in a permanent file which shall be readily available for audit. This is included in COL Action Item 3.10-1. The staff may audit these files to review the results of tests and analyses that were performed to (1) ensure that the criteria in the CESSAR-DC were properly implemented, (2) ensure that adequate qualification was demonstrated for all equipment and their supports, and (3) verify that all applicable loads were properly defined and accounted for in the testing and analyses performed.

To follow the 10 CFR Part 52 design certification process, any change to the commitments involving seismic and dynamic qualification of mechanical and electrical equipment discussed in Section 3.10 of this report would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Furthermore, any requested change to these commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

On the basis of its review of CESSAR-DC Sections 3.9.2.2, 3.9.3.2, and 3.10, as set forth above, the staff concludes that ABB-CE has defined appropriate seismic

and dynamic qualification programs for mechanical and electrical equipment, and has also specified appropriate pump and valve operability assurance programs. These programs meet applicable portions of GDC 1, 2, 4, 14, and 30, Appendix B to 10 CFR Part 50, 10 CFR 52.47, Appendix A to 10 CFR Part 100, and SECY-93-087, and are acceptable. The conclusion is based on the following.

The qualification program which will be implemented for mechanical, instrumentation, and electrical equipment meets the requirements and recommendations of IEEE 344-1987 and the regulatory positions of RGs 1.61, 1.89, 1.92, 1.100, and SRP Section 3.9.3, and provides adequate assurance that such equipment will function properly under all imposed design and service loads, including the loadings imposed by the SSE, postulated accidents, and LOCAs. This program constitutes an acceptable basis for satisfying the applicable requirements of GDC 2, 4, 14, and 30 of Appendix A to 10 CFR Part 50 and Appendix A to 10 CFR Part 100 as they relate to qualification of equipment.

On this basis, DSER Open Item 3.10-1 is resolved.

3.11 EQ of Mechanical and Electrical Equipment

3.11.1 Introduction

Equipment that is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life, for the time it is required to operate. This requirement, which is embodied in GDC 1 and 4 of Appendix A to 10 CFR Part 50 and Criteria III, XI, and XVII of Appendix B to 10 CFR Part 50, is applicable to equipment located inside and outside the containment. More detailed requirements and guidance related to the methods and procedures for demonstrating this capability for electrical equipment are in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which supplements IEEE 323 and various RGs and industry standards, and RG 1.89, Revision 1.

3.11.2 Background

The staff issued NUREG-0588 in December 1979 to promote a more orderly and systematic implementation of equipment qualification programs by industry and to guide the staff in its use in ongoing licensing reviews. The positions in NUREG-0588 provide guidance on (1) how to establish EQ service conditions, (2) how to select methods

that are considered appropriate for qualifying equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation. A final rule on EQ of electrical equipment important to safety for nuclear power plants became effective on January 21, 1983. This rule, 10 CFR 50.49, specifies the requirements for demonstrating the EQ of electrical equipment important to safety that is located in harsh environments. Each item of electric equipment important to safety must be qualified by one of the following methods: (1) testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable, (2) testing a similar item of equipment with a supporting analyses to show that the equipment to be qualified is acceptable, (3) experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable and (4) analysis in combination with partial type test data that supports the analytical assumptions and conclusions. In RG 1.89, Revision 1 (June 1984), the staff specifies guidelines for complying with the rule. The applicant or licensee shall prepare a list of electrical equipment important to safety covered by the qualification requirements. In addition, the applicant or licensee shall include the following information for electric equipment important to safety in a qualification file: (1) the performance specifications under conditions existing during and following design basis accidents, (2) the voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with (1) above can be ensured, and (3) the environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with (1) and (2) above. The applicant or licensee shall keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment is important to safety meets the requirements. In conformance with 10 CFR 50.49, electrical equipment for PWRs referencing the System 80+ standard design must be qualified according to the criteria in Category I of NUREG-0588 and RG 1.89, Revision 1.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B to 10 CFR Part 50. The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment.

To document the degree to which the EQ program for the System 80+ standard design complies with the EQ requirements and criteria, ABB-CE submitted

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CESSAR-DC Section 3.11, "Environmental Design of Mechanical and Electrical Equipment," and CESSAR-DC Appendices 3.11A and 3.11B, "Typical Environmental Conditions and Test Profiles for Structures and Components" and "Identification, Location, and Typical Environmental Condition of Equipment," respectively, and responded on February 12, 1992 (LD-92-017), to a staff RAI dated October 10, 1991.

3.11.3 Staff Evaluation

The staff limited its evaluation of the EQ program for the System 80+ standard design to a review of ABB-CE submittals on its approach for selecting and identifying equipment required to be environmentally qualified for the System 80+ standard design, qualification methods proposed, and completeness of information in the tables in CESSAR-DC Appendices 3.11A and 3.11B. The bases for the staff's evaluation are SRP Section 3.11, Revision 2; NUREG-0588, Category 1; RG 1.89, Revision 1; and 10 CFR 50.49. For COL applicants referencing the System 80+ certified design, the staff will review specific details of the EQ programs for their plants using the evaluation bases mentioned above.

3.11.3.1 Completeness of Qualification of Electrical Equipment Important to Safety

The following three categories of electrical equipment important to safety must be qualified in accordance with the provisions 10 CFR 50.49(b)(1), (b)(2), and (b)(3):

- (b)(1) – safety-related electrical equipment (relied on to remain functional during and after design-basis events)
- (b)(2) – non-safety-related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory performance of the safety functions by the safety-related equipment
- (b)(3) – certain postaccident monitoring equipment (Categories I and II postaccident monitoring equipment as specified in RG 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident").

CESSAR-DC Appendix 3.11B lists the equipment required to mitigate a DBA or to attain a safe shutdown. The CESSAR-DC states that specific equipment for each system is discussed in the appropriate section of the CESSAR-DC as referenced by Appendix 3.11B. The CESSAR-DC also states that the design of the information systems important to safety conform to the guidelines of Revision 3 of RG 1.97. However, the footnote for

10 CFR 50.49(b)(3) references Revision 2 for selection of the types of postaccident monitoring equipment. In issuing Revision 3, the NRC staff stated that conformance to Revision 3 would not alter the implementation for 10 CFR 50.49. In accordance with 10 CFR 50.12(2)(ii), the Commission will consider granting an exemption if application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. Therefore, conformance to Revision 2 is not required because conformance to Revision 3 meets the underlying purpose of the rule. Therefore, an exemption from 10 CFR 50.49(b)(3) is justified by the special circumstances set forth in 10 CFR 50.12 (a)(2)(ii). On the basis of its review of CESSAR-DC Appendix 3.11B, the staff finds ABB-CE's approach for selecting electrical equipment required to be environmentally qualified acceptable. The staff will review specific details provided by applicants referencing the System 80+ certified design to demonstrate their compliance with 10 CFR 50.49(b)(1), (b)(2), and (b)(3) with respect to identification of electrical equipment important to safety required to be environmentally qualified. The details must list systems and their components that are included in the plant EQ program, and design features for preventing the potential adverse consequences identified in IE Information Notice 79-22, "Qualification of Control Systems."

ABB-CE has elected to use the new accident source term described in draft NUREG-1465. The staff's acceptance of the new accident source term for evolutionary designs, such as CESSAR-DC System 80+, is discussed in Appendix 15.A.1 of this report.

The radiation qualifications for individual safety-related components are developed on the basis of two conditions:

- the radiation environment expected at the component location from equipment installation to the end of qualified life, including the time the equipment is required to remain functional after the accident
- the limiting design-basis accident for which the component provides a safety function

These design-basis accidents are divided into two general classes: LOCA and non-LOCA events. For non-LOCA events, such as main steam and feedwater line breaks, the source terms are developed from conservative estimates of fuel assembly gap activity releases, as discussed in Section 15.A.2 of this report and in CESSAR-DC Chapter 15, and the maximum reactor coolant specific activities, as discussed in CESSAR-DC Section 11.1. For LOCA

events, the maximum reactor coolant specific activities, as discussed in CESSAR-DC Section 11.1, are also included, but two levels of fuel damage are considered.

One level corresponds to a design-basis accident, without significant fuel melting, involving decay power heatup of the fuel assemblies to a point at which 100 percent of the gap activity is released. The timing of the coolant and gap activity releases is taken from draft NUREG-1465 with one exception; that is, 20 percent of the gap activity is assumed to be released instantaneously rather than according to the uniform release rate assumed in draft NUREG-1465. Components needed to preclude or limit core damage must be qualified. Within this category are the EFWS and the components identified in List 1 of CESSAR-DC Table 3.11-2.

The second level of fuel damage is the level corresponding to 10 CFR Part 100 which calls for a postulated design-basis accident involving substantial melting of the fuel. The timing of the coolant and gap activity releases is taken from draft NUREG-1465. Qualification is required for components needed to maintain containment integrity, including the long-term cooling of in-vessel debris. In this category is all other equipment within the scope of 10 CFR 50.49. The long-term cooling of the in-vessel debris does not require the EFWS. However, to ensure proper transition from arresting core damage to the long-term cooling mode, the EFWS is qualified to the 10 CFR Part 100 DBA source term for a period of 72 hours.

3.11.3.2 Qualification Methods

3.11.3.2.1 Electrical Equipment in a Harsh Environment

Detailed procedures for qualifying safety-related electrical equipment located in a harsh environment are defined in NUREG-0588 and RG 1.89. The criteria in these documents are also applicable to other equipment important to safety defined in 10 CFR 50.49.

The methodology used by ABB-CE to qualify NSSS safety-related electrical equipment is outlined in ABB-CE's Topical Report CENPD-255-A, Revision 3, dated October 1985, entitled "Qualification of Class 1E Electrical Equipment." This topical report is a generic document that has been reviewed and approved by the NRC staff by letter dated August 13, 1985, entitled Acceptance For Referencing of Licensing Topical Report CENPD-255-A Revision 3, "Class 1E Qualitification - Qualification of Class 1E Electrical Equipment." The staff's approval of CENPD-255-A, Revision 3, incorporated Amendment 9 of CESSAR-F (Combustion Engineering, "System 80 - Combustion Engineering Standard Safety Analysis Report,

Final Safety Analysis Report," Amendment 9, February 1984). The staff's review of CESSAR-DC included a review of the basis for incorporating Amendment 9 of CESSAR-F into the review of CENPD-255-A, Revision 3. ABB-CE stated that referencing CENPD-255-A, Revision 3, is independent of references to Amendment 9 of CESSAR-F. Consequently, the NRC staff concludes that it is acceptable for CESSAR-DC to reference CENPD-255-A Revision 3, without reliance upon Amendment 9 of CESSAR-F, and that where CENPD-255-A and CESSAR-DC differ, CESSAR-DC takes precedence. COL applicants referencing CENPD-255-A, Revision 3, must provide actual, plant-specific values that are in compliance with the requirements of 10 CFR 50.49. In addition, each COL applicant must verify the completeness of the equipment lists through use of the appropriate interface criteria and the validity of the environmental classification for each equipment item. On the basis of the staff review and approval of CENPD-255-A, Revision 3, the staff finds the methodology acceptable.

In Amendment I to CESSAR-DC Section 3.11.2, ABB-CE proposed an alternative test profile which allowed testing twice at the specified service condition as a substitute for testing once at a profile that includes margins. In the DSER, the staff stated that this approach is nonconservative; not consistent with IEEE 323-1974, NUREG-0588, or RG 1.89; and does not meet the intent of nor is it in compliance with 10 CFR 50.49. Consequently, this was identified as DSER Open Item 3.11.3.2.1-1. Subsequently, in Section 3.11.2 of Amendment N to CESSAR-DC, ABB-CE stated that environmental qualification of electrical equipment located in harsh environments within ABB-CE's scope of supply will be in accordance with the methodologies outlined in CENPD-255-A, Revision 3. ABB-CE also stated that environmental qualification of electrical equipment outside of ABB-CE's scope of supply will be in accordance with IEEE 323-1974 and RG 1.89, Revision 1. CENPD-255-A, Revision 3, and IEEE 323-1974 do not include an alternate test profile which allows for the substitution of testing twice at the specified service condition as a substitute for testing once at a profile which includes margins. This is acceptable and resolves DSER Open Item 3.11.3.2.1-1.

To date, the NRC staff has not endorsed IEEE 323-1983; therefore, references to this standard in its entirety, or in part, are not acceptable. In Amendment I to CESSAR-DC Section 3.11.2.1, IEEE 323-1983 was referenced in regard to the definition of a mild environment. This was DSER Open Item 3.11.3.2.1-2. Subsequently, in CESSAR-DC Section 3.11.2.1, IEEE 323-1983 is no longer referenced in relation to EQ of electric equipment important to safety for nuclear power plants. This is acceptable. On this basis, DSER Open Item 3.11.3.2.1-2 is resolved. An

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acceptable definition for a mild environment is in 10 CFR 50.49, paragraph (c). Resolution of this open item also resolves the staff's concerns related to the definition of a mild environment for mechanical equipment.

In addition, for current-generation operating reactors, the staff's definition of what constitutes a mild radiation environment for electronic components such as semiconductors, or any electronic component containing organic materials, is different from what it is for other equipment. The staff position is that a mild radiation environment for electronic equipment is a total integrated dose of less than 10 Gy (10^3 Rad). For other equipment it is less than 10^2 Gy (10^4 Rad). With the expected significant increase in the quantity and variety of electronic components in newer generation plants, the staff has increasing concerns about the efforts being made and the ability of these components to be environmentally qualified. In the DSER, the staff stated that ABB-CE should address the staff's concerns about the EQ of electronic components. This was identified as DSER Open Item 3.11.3.2.1-3. In CESSAR-DC Amendment Q, Section 3.11.2.2, ABB-CE stated that equipment which is exposed to radiation equal to or above 10^2 Gy (10^4 Rad) (equal to or above 10 Ry (10^3 Rad) for electronic equipment) will be irradiated to its anticipated total integrated dose prior to type testing, unless determined by analysis that radiation does not affect its ability to perform its required function. This is acceptable and resolves DSER Open Item 3.11.3.2.1-3.

Topical Report CENPD-255-A, Revision 3, was written by ABB-CE and reviewed and approved by the NRC staff for equipment within ABB-CE's scope of supply. However, in Amendment I to CESSAR-DC Section 3.11.2 and in ABB-CE's submittal of February 12, 1992 (LD-92-017), ABB-CE stated that the acceptance of this report should be extended to include other equipment suppliers. The staff did not agree with this position; therefore, this was identified as DSER Open Item 3.11.3.2.1-4. In Amendment N to CESSAR-DC Section 3.11.2, ABB-CE stated that EQ of electrical equipment outside of ABB-CE's scope of supply will be in accordance with IEEE 323-1974 and RG 1.89, Rev. 1. This is acceptable and resolves Open Item 3.11.3.2.1-4.

As discussed in Section 3.11.3.1 above, ABB-CE elected to use the new accident source term described in draft NUREG-1465. In Section 15.A.2 of this report, the staff determined that the appropriate application of the source term in draft NUREG-1465 should be based on the use of gap release and the early in-vessel releases for design-basis accident evaluations as proposed by ABB-CE. Additionally, the staff determined it appropriate that the qualification of individual components be based on the limiting design-

basis accident for which the component provides a safety function. Therefore, the ABB-CE method for determining the radiation qualifications of components within the scope of 10 CFR 50.49 is acceptable.

3.11.3.2.2 Safety-Related Mechanical Equipment in a Harsh Environment

Although no detailed requirements exist for mechanical equipment, GDC 1 and 4 and Appendix B to 10 CFR Part 50 (Criteria III, "Design Control," and XVII, "Quality Assurance Records") contain the following requirements related to equipment qualification:

- Components should be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.
- Measures should be established for the selection and review for the suitability of application of materials, parts, and equipment that are essential to safety-related functions.
- Design control measures should be established for verifying the adequacy of design.
- Equipment qualification records should be maintained and should include the results of tests and materials analyses.

For mechanical equipment, the staff concentrates its review on materials that are sensitive to environmental effects, for example, seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms. A review and evaluation should be done to

- identify safety-related mechanical equipment located in harsh environment areas, including required operating time
- identify non-metallic subcomponents of this equipment
- identify the environmental conditions for which this equipment must be qualified (The environments defined in the electrical equipment program are also applicable to mechanical equipment.)
- identify non-metallic material capabilities
- evaluate environmental effects

In LD-92-017, ABB-CE proposed a revision to CESSAR-DC Sections 3.11.2.1 and 3.11.3.2 to provide additional clarifying information and committed the EQ of

mechanical equipment to CENPD-255-A, Revision 3. This proposed change was intended to resolve the staff concerns with this section. Therefore, in the DSER, this was identified as DSER Confirmatory Item 3.11.3.2.2-1. The staff noted that ABB-CE should incorporate the proposed changes into CESSAR-DC Sections 3.11.2.1 and 3.11.3.2. Subsequently, ABB-CE revised CESSAR-DC Sections 3.11.2.1 and 3.11.3.2 in Amendments N and Q, respectively, to eliminate reference to IEEE 323-1983 for a mild environment and to add reference to CENPD-255-A, Revision 3 for EQ of mechanical equipment. This is acceptable and resolves DSER Confirmatory Item 3.11.3.2.2-1.

3.11.3.3 Conclusions

On the basis of its review of the CESSAR-DC, other applicant submittals, and previous review and acceptance of CENPD-255-A, Revision 3, the staff concludes that the program proposed by ABB-CE for environmentally qualifying electrical equipment important to safety and safety-related mechanical equipment is acceptable.

By DSER COL Action Item 3.11.3.3-1, the staff noted that for applicants referencing the System 80+ certified design, the specific details of their plant-specific EQ program will be made available to the staff, including the results of maintenance and surveillance, for applicable equipment located in potentially harsh environmental zones. Subsequently, CESSAR-DC Section 3.11 incorporates the above action. This is acceptable. This is included in COL Action Item 3.11-1.

3.12 Piping Design

3.12.1 Introduction

This section is the staff's safety evaluation of ABB-CE's design acceptance criteria (DAC) for the System 80+ piping design. The staff used the SRP guidelines to evaluate the piping design information in the CESSAR-DC and performed a detailed audit of the piping design criteria, including sample calculations. The staff evaluated the adequacy of the structural integrity and functional capability of safety-related piping systems. The review was not limited to the ASME Code Class 1, 2, and 3 piping and supports, but included buried piping, instrumentation lines, the interaction of non-Category I piping with seismic Category I piping, and any safety-related piping designed to industry standards other than the ASME Code. The staff's evaluation of the adequacy of the System 80+ piping design analysis methods, design procedures, acceptance criteria, and related ITAACs that are to be used for the completion and verification of the System 80+

piping design appears in the following sections of this report. The staff's evaluation includes

- applicable codes and standards
- methods to be used for completing the piping design analyses
- modeling techniques
- pipe stress analyses criteria
- pipe support design criteria
- criteria for postulating high-energy line breaks
- LBB approach applicable to the System 80+ design
- generic piping design ITAACs

In order to approve the System 80+ design, the staff must arrive at a final determination that, if the COL applicant or licensee successfully completes the piping design and analyses and the ITAAC as required by 10 CFR Part 52 using the design methods and acceptance criteria discussed herein, there will be adequate assurance that the piping systems will perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

The staff's evaluation of the System 80+ piping DAC is based primarily upon the staff review of piping design and analysis-related information in CESSAR-DC Section 3.6, 3.7, and 3.9 and Appendix 3.9A, and upon the staff's audit of several sample piping analysis problems provided by ABB-CE. The sample problems included the pressurizer SL, the shutdown cooling system (SCS) line, and the feedwater economizer line.

3.12.2 Codes and Standards

3.12.2.1 ASME Code

In SECY-93-087, the staff recommended using the newest codes and standards that have been endorsed by the NRC. Revisions to codes and standards that have not been formally endorsed by the NRC will be reviewed on a case-by-case basis. The Commission approved the staff position in its SRM dated July 21, 1993. For the System 80+ design certification, ABB-CE has established that the ASME Code, Section III, will be used for the design of ASME Code Class 1, 2, and 3 pressure retaining components and their supports. The specific edition and addenda are listed in CESSAR-DC Table 1.8-6. The ASME Code is considered Tier 1 information; however, the specific edition and addenda are considered Tier 2 information. The specific edition and addenda are considered as Tier 2 information because of the continually evolving technical nature associated with the design and construction practices (including inspection and examination techniques) of the code. Fixing a specific edition and specific addenda

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during the design certification stage would result in inconsistencies between design and construction practices during the detailed design and construction stages. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although the reference to a specific edition of the code for the design of ASME Code class components and their supports is sufficient to reach a safety finding during the design certification stage, it is necessary that the construction practices and examination methods of an updated code that would be effective at the COL stage be consistent with the design practices established at the design certification stage.

The staff finds that the specification of the ASME Code as Tier 1 information and the specific edition and addenda as Tier 2 information is appropriate because it would provide the COL applicant the means to revise or supplement the referenced code edition with portions of the later code editions and addenda needed to ensure consistency between the design for the System 80+ pressure-retaining components and their supports and construction practices. In this manner, the updated reference code to be used at the time of the COL application is ensured to be consistent with the latest design, construction, and examination practices at that time. However, where the staff finds that there might be a need to specify certain design parameters from a specific code edition or addenda during its design certification review, particularly when that information is of importance to establish a significant aspect of the design or is used by the staff to reach its final safety determination, such considerations, if necessary, are reflected in the applicable sections of this safety evaluation.

Therefore, all ASME Code Class 1, 2, and 3 pressure retaining components and their supports shall be designed in accordance with the requirements of ASME Code, Section III, using the specific edition and addenda named in the CESSAR-DC. The COL applicant may opt for later editions and addenda of the code to ensure that the design is consistent with the construction practices (including inspection and examination methods) in effect at the time of application for the COL. The portions of the later code editions and addenda shall be presented to the NRC staff for review and approval along with the COL application.

3.12.2.2 ASME Code Cases

The only acceptable ASME Code cases that may be used for the design of ASME Code Class 1, 2, and 3 piping systems in the System 80+ standard plant are those either conditionally or unconditionally endorsed in RGs 1.84 and 1.85 in effect at the time of design certification or specifically reviewed and approved by the staff as listed below. However, the COL applicant may submit with its COL

application, for staff review and approval, future code cases that are endorsed in RGs 1.84 and 1.85 (at the time of COL) provided they do not alter the staff's safety findings on the System 80+ certified design.

In CESSAR-DC Table 1.8-7 of the early amendments, ABB-CE submitted a list of ASME Code Cases for use in System 80+ piping and pipe support design. After discussion with the staff, ABB-CE presented the following list in the CESSAR-DC:

- ASME Code Case N-71-15, "Additional Materials for Subsection NF, Classes 1, 2, 3, and MC Component Supports Fabricated by Welding, Section III, Division 1." This code case has been endorsed by the staff subject to the additional conditions given in RG 1.85.
- ASME Code Case N-122-1, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 1 Piping, Section III, Division 1." Previously, this code case was endorsed by the staff in RG 1.84. However, endorsement of this edition was withdrawn due to obvious printing discrepancies. The staff's approval of this code case was contingent upon correction of these typographical errors.
- ASME Code Case N-192-2, "Use of Braided Flexible Connectors, Class 2 and 3." This code case has been endorsed by the staff subject to the additional conditions given in RG 1.84.
- ASME Code Case N-247, "Certified Design Report Summary for Component Standard Supports, Section III, Division 1, Classes 1, 2, 3 and MC." This code case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-249-10, "Additional Materials for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated Without Welding." This code case has been endorsed by the staff subject to the additional conditions given in RG 1.85.
- ASME Code Case N-262, "Resistance Spot Welding for Structural Use in Component Supports." This code case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-309-1, "Identification of Materials for Component Supports." This code case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-313, "Alternate Rules for Half-Coupling Branch Connections, Class 2." This code case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-318-4, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1." This code case has been endorsed by the staff subject to the additional conditions given in RG 1.84. This is further discussed in Section 3.12.5.16 of this report.
- ASME Code Case N-319-1, "Alternate Procedure for Evaluation of Stresses in Butt Weld Elbows in Class 1 Piping, Section III, Division 1." This code case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-391-1, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1." This code case has been endorsed by the staff subject to the additional conditions in RG 1.84.
- ASME Code Case N-392-1, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1." This code case has been endorsed by the staff subject to the additional conditions in RG 1.84. This is further discussed in Section 3.12.5.16 of this report.
- ASME Code Case N-393, "Repair Welding Structural Steel Rolled Shaped and Plates for Component Supports, Section III, Division 1." This code case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis for Class 1, 2, and 3 Piping." This code case has been endorsed by the staff subject to the additional conditions in RG 1.84. This is further discussed in Section 3.12.5.4 of this report.
- ASME Code Case N-420, "Linear Energy Absorbing Supports for Subsection NF, Classes 1, 2, and 3 Construction, Section III, Division 1." This code case has been endorsed by the staff subject to the additional conditions given in RG 1.84.
- ASME Code Case N-430, "Alternative Requirements for Welding Workmanship and Visual Acceptance Criteria for Class 1, 2, 3 and MC Linear-Type and Standard Supports." This code case has been endorsed by the staff subject to the additional conditions given in RG 1.84.
- ASME Code Case N-433, "Non-threaded Fasteners for Class 1, 2, and 3 Component Piping Supports." This code case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-474-1, "Design Stress Intensities and Yield Strength Values for UNS NO6690 with a minimum specific Yield Strength of 35 ksi, Class 1 Components." This code case has been endorsed by the staff in RG 1.85.
- ASME Code Case N-476, "Class 1, 2, 3, and MC Linear Component Supports - Design Criteria for Single Angle Members, Subsection NF." This code case has been endorsed by the staff in RG 1.84.

The staff concludes that because all of these code cases either meet the guidelines of RGs 1.84 or 1.85, or have been reviewed and endorsed by the staff, they are acceptable for use in the System 80+ design.

3.12.2.3 Design Specifications

ASME Code, Section III, requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design data inputs. The Code also requires a design report for ASME Code Class 1, 2, and 3 piping and components. In the CESSAR-DC, ABB-CE commits to construct all safety-related components, such as vessels, pumps, valves, and piping systems to applicable requirements of the ASME Code, Section III.

During its review of the CESSAR-DC, the staff noted that although it is understood that design reports will not be available at the time of certification, design specifications, or at least the methodologies for preparing them, should be available since design specifications serve as the basis for construction. In the absence of preparing actual design specifications at the time of design certification, the staff asked ABB-CE to prepare a document that discusses the requirements and methodologies for preparing design specifications that should be followed by ABB-CE or any applicant for referencing the System 80+ design. In response to the staff request, ABB-CE submitted a document entitled, "Procedure Guideline for Preparing the ASME Code, Section III Design Specification for the System 80+ Certified Design." The staff reviewed this document and determined that it provides adequate guidelines for preparing design specifications and contains sufficient details required by the ASME Code, and is thus acceptable.

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3.12.2.4 Conclusions

The staff finds that in CESSAR-DC Section 3.9.3, ABB-CE meets the requirements for and commitments to the applicable codes and standards contained in 10 CFR 50.55a and GDC 1 as they pertain to the codes and standards specified for ASME Code, Class 1, 2, and 3 components, by ensuring that systems and components important to safety are designed to quality standards commensurate with their importance to safety.

3.12.3 Analysis Methods

The staff reviewed the information in CESSAR-DC Section 3.9.1 relative to the design transients and methods of analysis used for all seismic Category I piping and pipe supports designated as ASME Code Class 1, 2, and 3 under ASME Code, Section III, as well as those not covered by the Code. It reviewed the assumptions and procedures used for the inclusion of transients in the design and fatigue evaluation of ASME Code Class 1 and core support components. It also reviewed the computer programs used in the design and analysis of seismic Category I components and their supports, as well as experimental and inelastic analytical techniques.

3.12.3.1 Experimental Stress Analysis

During an audit performed on June 23 through 25, 1993, at ABB-CE offices in Windsor, Connecticut, ABB-CE was asked to specify any piping components for which experimental stress analysis would be performed. ABB-CE responded that it is not planning to use experimental stress analysis in piping design. If a COL applicant wishes to use this method in any System 80+ piping design, the details of the method as well as the scope and extent of its application should be submitted to the staff for approval before its use. The staff's position is that experimental stress analysis methods shall be in compliance with Appendix II of the ASME Code, Section III, Division I.

3.12.3.2 Modal Response Spectrum Method

Static and dynamic analyses of all major seismic Category I piping systems and components are based on linear-elastic analysis methods. Seismic analysis of a piping system is generally performed using both dynamic and static analysis techniques. A dynamic analysis is performed to evaluate the inertia loads developed as the mass of the piping is accelerated by seismic motion. A static analysis is performed to determine loading resulting from differential seismic movements of structures or large lines

to which piping is attached. The dynamic analysis of a piping system is generally performed by the modal response spectrum method of analysis. Alternate analysis methods include the time-history method and the equivalent static method.

CESSAR-DC Section 3.7.3 and Appendix 3.9A, Section 1.4.3.2.1, describe the piping dynamic analysis procedure using the modal response spectrum method. First a mathematical model is constructed to reflect the dynamic characteristics of the piping system. The model consists of lumped masses connected by elastic members. As a minimum, the number of DOFs is equal to twice the number of modes with frequencies less than the frequency corresponding to the ZPA. The stiffness matrix is determined from the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations, as well as changes in stiffness due to curved members. The mode shapes and natural frequencies for all significant modes of vibration are computed. For each direction of earthquake motion, modal participation factors for all significant modes are calculated. Using the appropriate response spectrum, the spectral acceleration for each mode is determined. For a piping system supported at points with different dynamic excitations, an enveloped response spectrum of all attachment points is used. From the mode shapes, participation factors, and spectral accelerations of individual modes, the modal responses are calculated. They include the modal inertia response forces, moments, displacements, and accelerations. For a given direction, the modal responses are combined in accordance with the methods described in RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 1.

The modal response is calculated for each of the three earthquake directions (Two horizontal and the vertical). The total seismic response from the simultaneous application of the 3-D components of earthquake loading are obtained by combining the maximum codirectional responses to each of the three components by the SRSS method.

The effects of SAM are considered by performing a static analysis which accounts for the relative displacement effects between support locations. For models with piping in more than one building, it is assumed that the buildings move 180 degree out of phase. Within the RB, there are differential movements between the RB, the containment vessel, reactor interior structures, and the NSSS. These movements are assumed to act 180 degree out of phase. The resulting relative movements are applied as static support displacements with all dynamic supports active. The resulting stresses are placed in the secondary stress category because they are displacement-induced and self-limiting. The results of the static analysis are combined

with the results of the response spectrum analysis by absolute summation to determine the total seismic response of the piping.

The staff reviewed the CESSAR-DC description of the modal response spectrum method and found that it is consistent with the applicable guidelines in SRP Section 3.9.2 and is, therefore, acceptable.

3.12.3.3 Independent Support Motion Method

The independent support motion (ISM) response spectrum analysis method is an alternate analysis method which can be used for piping systems that are supported at points with different dynamic excitations. When this method is used, the responses caused by motions of supports between two or more support groups may be combined by the SRSS method if a support group is defined by supports that have the same time-history input. This usually means all supports located on the same floor, or positions of a floor or a structure. In addition, this method should be implemented in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, Volume 4.

In CESSAR-DC Section 3.7.3.9 of the early amendments, ABB-CE stated that when the equipment or component is supported at points with different elevations, either the envelope of the response spectra at a given elevation or a multiple support excitation approach is used for the seismic qualification of the equipment. For the multiple support excitation, a time-history analysis method or ISM response spectrum method is used. During an audit performed on April 6-8, 1993 at DE&S offices in Charlotte, North Carolina, ABB-CE was asked to submit a sample piping analysis using the ISM response spectrum method if this methodology will be used in System 80+ piping design. At a follow-up audit at the Duke office on May 18-19, 1993, ABB-CE informed the staff that, upon further review, it decided that the ISM response spectrum analysis method will not be used in System 80+ piping design.

On the basis of the information given during the audits, the staff requested that CESSAR-DC Section 3.7.3.9 be revised to state that the ISM response spectrum analysis method will not be used for piping analysis. Subsequently, ABB-CE revised CESSAR-DC Section 3.7.3.9 by Amendment Q, to permit only the time-history analysis method for the multiple support excitation. The staff finds this revision acceptable. If a COL applicant wishes to use the ISM response spectrum analysis method in any System 80+ piping design, a detailed description of the methodology containing a sample analysis and a description

of the computer code and its verification shall be submitted to the staff for approval before its use.

3.12.3.4 Time-History Analysis Method

The time-history analysis method may be used to determine the dynamic response of piping systems. For the System 80+ piping design, ABB-CE uses this method as an alternative to the response spectrum method for seismic analysis. The method is also used for dynamic analyses of other types of load such as LOCAs or hydraulic transients. This method uses either a direct integration or modal superposition method to solve the equations of motion.

CESSAR-DC Section 3.7.2.1.1.2 and Appendix 3.9A, Section 1.4.3.2.2, describe the piping analysis procedure using the time-history analysis method. The development of mathematical models which define flexibility and mass and the calculation of natural frequencies and mode shapes are performed in the same manner as described for response spectrum analysis. The solution of the equations of motion is obtained by either modal superposition or direct integration.

The modal superposition method is used when the equations of motion can be decoupled. The decoupled equation of motion for each mode is integrated and the total response is obtained by algebraically summing up the responses of the individual modes. The principle of superposition holds as long as the system is linear. When this method is used, the number of modes analyzed is selected to account for the principal vibration modes of the system based upon mass and stiffness properties, modal participation factors and closeness to amplified region of the response spectrum of the input time history. As required on a case-by-case basis, the analysis is repeated with more modes in order to verify the adequacy of the chosen number of modes.

When the direct integration method is used, the solution of the eigenvalue problem is not required. Direct integration of the equations of motion by either implicit or explicit methods of numerical integration is used to solve the equations of motion. The integration time step is chosen to be a maximum of 1/10 of the period of the highest frequency of interest for the system. As required on a case-by-case basis, the size of time step for integration is reduced and the analysis repeated in order to verify the choice of the time step. In addition, to account for uncertainties, the time step size is adjusted by ± 15 percent to achieve the frequency band-broadening effects. When this method is used, viscous damping proportional to the mass and stiffness matrices is used. The values of α and β constants are selected so that the damping in the range

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of frequency of interest is approximately equal to the damping of the system.

The piping model is subjected to seismic excitations at the anchor points and at the building supports. For statistically independent earthquake motions, input excitations in all three translation directions and, as applicable, in all three rotational directions, maybe applied simultaneously to the anchor points and building supports.

Input of multiple-support time-history excitations, which allow calculation of the effects of both differential motion and inertia, may be used in a piping system. An alternative time-history method is to use an "envelope" time history excitation to calculate the inertia response and to perform a separate static analysis to determine the effects of differential support motion. The envelope excitation is a time history whose response spectrum envelopes the response spectra for the individual support motions. This method is described in ASME Code, Section III, Division 1, Appendix N, Section N-1228.4.

The staff reviewed the CESSAR-DC descriptions of the modal superposition and the direct integration time-history analysis methods and found them to be in compliance with the applicable guidelines of SRP Section 3.9.2 and, therefore, acceptable.

3.12.3.5 Inelastic Analysis Method

The CESSAR-DC does not contain any information on the use of inelastic analysis methods for the System 80+ piping. During an audit performed on June 23 through 25, 1993, at the ABB-CE office in Windsor, Connecticut, ABB-CE was asked to identify any piping components for which inelastic analysis methods would be used. ABB-CE stated that it was not planning to use inelastic analysis methods in piping design. If inelastic methods are to be used in any System 80+ piping analysis, the details of the inelastic method and its acceptance criteria, as well as the scope and extent of its application, should be submitted to the staff for approval before its use by a COL applicant.

3.12.3.6 Small-Bore Piping Method

The early versions of the CESSAR-DC did not provide any specific information regarding the method for the structural design of small-bore piping systems in the System 80+ plant. During the audit on June 23 through 25, 1993, at the ABB-CE office in Windsor, Connecticut, ABB-CE was asked to submit a description of the small-bore piping design method in the CESSAR-DC. Subsequently, ABB-CE submitted CESSAR-DC Appendix 3.9A, Section 1.1, which specifies the analysis requirements de-

scribed in this appendix apply to both large and small-bore piping. This is acceptable.

3.12.3.7 Non-Seismic/Seismic Interaction (II/I)

All non-seismic Category I piping (or other systems and components) should be isolated from seismic Category I piping. This may be achieved by designing a seismic constraint or barrier or by locating the two sufficiently apart to preclude any interaction. If it is impractical to isolate the Category I piping system, the adjacent non-seismic Category I system should be evaluated to the same criteria as the seismic Category I system.

For non-seismic Category I piping systems attached to seismic Category I piping systems, the dynamic effects of the non-seismic Category I system should be considered in the analysis of the Category I piping. In addition, the non-seismic Category I piping from the attachment point to the first anchor should be evaluated to ensure that, under all loading conditions, it will not cause a failure of the seismic Category I piping system. CESSAR-DC Section 1.5.2.3 of Appendix 3.9A and Section 3.7.3.13 contain criteria and methods for designing the region of a seismic/non-seismic piping interface in order to isolate the seismic response of the non-seismic piping from the seismic piping. The criteria and methods are consistent with the above staff positions and the applicable portions of SRP Section 3.9.2 and RG 1.29, and are, therefore, acceptable.

3.12.3.8 Buried Piping

CESSAR-DC Section 3.7.3.12.1 outlines the criteria for the analysis of buried seismic Category I piping systems. It states that the structural integrity of the buried piping is evaluated by accounting for two primary effects of earthquake ground motions. They are the strains and associated stresses induced by soil motions caused by the passage of seismic waves and seismically induced differential movements of structures which the pipe enters or connects. Friction between the pipe and the surrounding soil may be considered using conservative estimates of the associated frictional forces. The differential movements at the entry or connection points are assumed to be out of phase. Equivalent static analysis uses the principles of beams on elastic foundations.

Buried piping experiences both axial and bending strains and associated stresses from the seismic wave effects, seismic differential movements and thermal expansion/contraction. High axial strains are caused mostly by seismic waves in conjunction with the soil friction. These axial strains are therefore included in the code stress check. CESSAR-DC Section 3.7.3.12.1.3 presents an alternative stress check equation for Level D SSE loading

in lieu of ASME Code Eq. 9 of NC/ND-3655. In this alternate equation, the range of bending plus axial stresses due to SSE loading is limited to $3S_h$. CESSAR-DC Section 3.7.3.12.1.4 presents modified Code Eqs. 10 and 11 for evaluating thermal expansion and contraction and for the combined effects of pressure, weight, sustained mechanical loads, and thermal expansion and contraction.

The staff reviewed the information presented in CESSAR-DC Section 3.7.3.12.1 and found the analysis methodology to be in conformance with the applicable guidelines given in SRP Section 3.9.2 as it relates to Category I buried piping systems, which states that the piping analysis should consider earthquake-induced inertia effects, surrounding soil resistance on piping movements, differential anchor motions, and local soil settlements. During the audit on June 23-25, 1993, at the ABB-CE office in Windsor, Connecticut, ABB-CE was also asked to provide additional justification for the alternative stress check equations given in CESSAR-DC Sections 3.7.3.12.1.3 and 3.7.3.12.1.4 as follows. The code equation on thermal expansion was modified to include additional consideration on axial stress, and the stress allowable for combined thermal and anchor motion effects did not conform with NRC guidelines on single earthquake design. In addition, ABB-CE was asked if buried piping will be installed in tunnels. If the pipe is not installed in tunnels, ABB-CE was asked to explain how the piping welds will be inspected.

In response to the staff's request, ABB-CE clarified CESSAR-DC Sections 3.7.3.12.1.3 and 3.7.3.12.1.4 to justify the alternative stress check equations as follows. In CESSAR-DC Section 3.7.3.12.1.4, ASME Code Eqs. 10 and 11 are modified to include the axial stresses due to thermal expansion/contraction frictional forces which are considered to be significant for buried pipes. Aside from this additional stress term, the equations are identical to the Code equations. The inclusion of the axial stress term into the Code equations is appropriate and the modified Eqs. 10 and 11 are acceptable for buried piping.

Subsequently, in CESSAR-DC Section 3.7.3.12.1.3, ABB-CE states that seismic loads on buried piping are considered secondary loads since they are generated by soil strains transmitted to the pipe and are, therefore, considered to be self-limiting. In addition, it further states that axial stresses are considered significant and are, therefore, included in the proposed Level D buried pipe stress check. The staff reviewed this information and agreed that there is a technical basis for treating the buried piping SSE stresses as secondary stresses and for including axial stresses in the stress check equation. However, the staff noted that the stress check equation should consider the combined range of moments and forces from all secondary

loads including thermal expansion. In addition, the staff also noted that ABB-CE had not submitted justification for a stress allowable of $3S_h$. Following additional discussions with the staff, ABB-CE modified, in CESSAR-DC Section 3.7.3.12.1.3, the Level D buried pipe stress check equation, which includes the range of resultant moments and forces due to the combination of SSE and thermal expansion/contraction. The stress from this load combination is limited to $3S_h$. The staff reviewed the modified equation and found that it is consistent with Eq. 10b from the NRC guidance document on single-earthquake design which limits the range of secondary stresses from the combined thermal expansion and SSE SAM to $3S_h$. In the case of buried piping, the total SSE stress is considered a secondary stress and is, therefore, included in the stress check equation. The staff, therefore, concludes that the modified Level D stress check equation is acceptable.

In response to the staff's questions on buried pipe inspections and tunnels, ABB-CE specified three piping systems which include buried pipe. They are the CCW system, the SSWS, and diesel generator engine fuel oil system. In CESSAR-DC Section 3.8.4.1.5, Amendment Q, ABB-CE stated that the CCW piping runs through a reinforced concrete tunnel below grade between the CCW Hx structure and the NA. The other two systems include piping that may be directly buried in the ground. ISI requirements for the SSWS are given in CESSAR-DC Section 9.2.1.4, Amendment R, in which ABB-CE stated that the SSWS is designed and installed to permit ISI and tests in accordance with ASME Code, Section XI. Buried SSWS piping is inspected by means of a leakage test that determines the rate of pressure loss or a test of differential flow between the ends of buried piping. ISI requirements for the diesel fuel oil system are given in CESSAR-DC Section 9.5.4.4, Amendment Q, in which ABB-CE stated that buried fuel oil system piping is inspected by means of a visual examination at each end of the buried piping (annulus enclosure) for evidence of leakage. The staff reviewed this information and determined that it provides an acceptable alternative to meet the inspection requirements of ASME Code, Section XI.

3.12.3.9 ASME Code, Section III, Appendix N

The staff has not endorsed the use of ASME Code, Section III, Appendix N, which is a nonmandatory appendix that is still evolving and does not currently agree with some regulatory positions. Therefore, for the System 80+ piping design, if the methodology in Appendix N is not consistent with regulatory positions discussed herein, the regulatory positions shall be used.

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3.12.3.10 Conclusions

On the basis of the evaluations in Section 3.12.3, the staff concludes that the analysis methods for all seismic Category I piping systems as well as non-seismic Category I piping systems that are important to safety are acceptable. The analysis methods use piping design practices that are commonly used in the industry and provide an adequate margin of safety to withstand the loadings as a result of normal operating, transient, and accident conditions.

3.12.4 Piping Modeling

3.12.4.1 Computer Codes

This section addresses the computer codes for analyzing piping systems in the System 80+ design. ABB-CE listed all computer programs it used for static and dynamic analyses to determine the structural and functional integrity of seismic Category I and non-seismic Category I items, in CESSAR-DC Section 3.9.1.2. The program summaries contained individual descriptions and applicability data. Design control measures to verify the adequacy of the design of safety-related components are required by Appendix B to 10 CFR Part 50. CESSAR-DC Section 3.9.1.2 states that the computer codes have been verified in conformance with design control methods consistent with the QA program described in Chapter 17 of the CESSAR-DC.

The staff performed an independent confirmatory piping stress analysis of representative piping systems in the System 80+ standard plant. These analyses verified the adequacy of the ABB-CE computer program used to generate the sample piping analyses that were audited by the staff on May 18 and 19, 1993, at the DE&S office in Charlotte, North Carolina. The confirmatory analysis verified that the ABB-CE computer program stated above is adequate with acceptable accuracy. On the basis of this evaluation, the staff concludes that the computer program verification process for the System 80+ piping design is acceptable.

3.12.4.2 Dynamic Piping Model

The requirements for the dynamic analysis of seismic Category I piping are in CESSAR-DC Section 3.7.3 and Appendix 3.9A. Each system is idealized as a mathematical model consisting of lumped masses connected by elastic members. A piping system is generally modeled as an assemblage of beams. The mass of each beam is lumped equally at its associated end nodes. Such concentrated weights as valves are modeled as lumped masses. The stiffness matrix for the piping system is determined using

the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations, as well as changes in stiffness as a result of curved members.

The staff reviewed the method for selecting the number of masses or DOF in the piping mathematical model to determine its dynamic response. In addition to the CESSAR-DC review, ABB-CE's piping design procedures and sample piping problems were audited by the staff on April 6 through 8, and May 18 and 19, 1993, at the office of DE&S in Charlotte, North Carolina. As a minimum, the number of DOFs is equal to twice the number of modes with frequencies less than the frequency corresponding to the ZPA. Pipe and fluid masses are lumped at nodes that are selected to coincide with the locations of large masses (e.g., valves, pumps, and motors) and with locations of significant geometric changes (e.g., pipe elbows, reducers, and tees). The torsional effects of valve operators and other equipment with offset center of gravity with respect to the centerline of the pipe are included in the model. In order to adequately represent the dynamic characteristics of the piping system, maximum mass point spacing shall not be greater than the length that would produce a natural frequency equal to the ZPA frequency when calculated based on a simply supported beam. Since this approach would, in effect, capture all modes up to the cutoff frequency, the staff finds that the System 80+ method for locating mass points is acceptable.

The staff reviewed the methodology for considering the effects of pipe support stiffness on the piping response. CESSAR-DC Appendix 3.9A, Section 1.7.2.8, states that supports are modeled in the piping analysis by using either the actual support stiffness values or by using rigid stiffness values. When the actual support stiffnesses are used, the flexibility of all support components, as well as the effects of the building structure, are included in the total stiffness value. Typically, rigid stiffness values are used for all but flexible supports (e.g., spring cans). When rigid stiffnesses are used, all supports in a given piping analysis are generally designed with reasonably equal stiffness to reduce the effects of load redistribution to stiffer supports due to the deflection of more flexible supports. The CESSAR-DC states that rigid supports are designed to ensure that the stiffnesses of the supports do not affect the pipe frequency. During the June 1993 staff audit, ABB-CE was asked to submit deflection limits for pipe supports that are modeled as rigid restraints. These limits were submitted on July 9, 1993, in a markup to revise CESSAR-DC Appendix 3.9A, Section 1.7.2.8. The revision required rigid support deflections to be limited to 1.6 mm (1/16 in.) for the greater of the SSE loadings or the minimum design loads of Section 7.2.13. In addition, the maximum deflection would be limited to 3.2 mm (1/8 in.) based on the maximum load combination. The staff

found these revisions acceptable. The deflection limit is consistent with good engineering practice to ensure a reasonable assumption that a pipe support is considered rigid in the piping analysis. The revision was subsequently incorporated into the CESSAR-DC, Amendment R. This is also discussed in Section 3.12.6.7 of this report.

Additionally, because the amplified response spectra are generally specified at discrete building node points, any additional flexibility between these points and the pipe support should be addressed. The additional response spectrum amplification may occur at equipment nozzles, flexible building steel, and at connections between piping runs and to branch lines. During the June 1993 staff audit, ABB-CE was asked to address this issue in the CESSAR-DC. In response, ABB-CE submitted markups to CESSAR-DC Appendix 3.9A with the following revisions. CESSAR-DC Appendix 3.9A, Section 1.4.3.2, was revised to state that piping systems will use appropriate response spectra or time-history input which includes amplification through auxiliary platforms or major building steel. ABB-CE revised CESSAR-DC Appendix 3.9A, Section 1.4.9, to state that response spectra are defined for NSSS equipment nozzles. For other equipment, response spectra are not defined. If the equipment natural frequencies are below the ZPA cutoff, the mass and stiffness properties are incorporated in the seismic analysis. ABB-CE revised CESSAR-DC Appendix 3.9A, Section 1.5.2.2, to give criteria for decoupling the seismic analysis of a branch line from the run line. When the decoupling criteria are met, the branch point is treated as an anchor in the analysis of the branch line. In that analysis, both the anchor movements and the inertial effects of the run pipe on the branch pipe will be considered where significant. The staff found these revisions acceptable. The markups were subsequently incorporated into the CESSAR-DC, Amendment R.

3.12.4.3 Piping Benchmark Program

To verify the adequacy of the computer program used by the COL applicant or licensee to complete the System 80+ piping system design and analyses, the NRC staff will establish mathematical models of representative piping systems in the System 80+ standardized plant to be used in a benchmark program. The mathematical models are based on the dynamic piping model and on the piping stress analysis criteria described in Section 3.12.4.2 and Section 3.12.5, respectively, of this report. The benchmark program verifies the adequacy of linear-elastic, dynamic piping analysis methods using the response spectrum method, modal superposition time-history method, and direct integration time-history method of analysis.

The benchmark program essentially consists of verifying mathematical models of the System 80+ pressurizer SL and feedwater economizer line using the COL applicant's or licensee's computer program. The piping configurations for the piping models are described in NUREG/CR-6128, "Piping Benchmark Problems for the ABB-CE System 80+ Standardized Plant," and considers piping dimensions, pipe sizes, materials, valve weights, support and anchor stiffnesses, and support locations. The piping input parameters for the benchmark analyses also are specified in the piping benchmark program and include damping values, loading definitions, and load combinations.

CESSAR-DC Section 3.9.1.2.1 states that computer codes for piping dynamic analysis will be benchmarked in accordance with NUREG/CR-6128. When the COL applicant's dynamic piping analyses are completed, the analysis results will be compared with benchmarked problems and evaluated, including the system modal frequencies, the maximum pipe moments, the maximum support loads and equipment reactions, and the maximum pipe deflections. The acceptance criteria or range of acceptable values are specified in the piping benchmark program and shall be satisfied. Any deviations from these values as well as the justification for such deviations shall be documented and submitted by a COL applicant or licensee to the NRC staff for review and approval before initiating final certified piping analyses.

The benchmark program ensures that the computer program used to complete the System 80+ piping design and analyses produces results that are consistent with results considered acceptable to the NRC staff.

3.12.4.4 Decoupling Criteria

When analyzing piping systems, the size of the mathematical model might exceed the capacity of the computer program if all large and small piping lines are included. Thus, small branch lines are generally decoupled from the large main piping. CESSAR-DC Appendix 3.9A, Section 1.5.2, provides criteria for the decoupling of the piping systems in the analysis model. It states that branch lines may be decoupled from the main run if the ratio of branch to run pipe nominal diameters is less than 0.33 or if the ratio of branch to run pipe moments of inertia is less than 0.04. In addition, branch lines must be designed flexible enough to absorb the anchor motions of the run pipe. The branch line flexibility is maintained by avoiding placement of branch line supports close to the run pipe. In the analysis, an appropriate stress intensity factor must be included on the branch and main run lines at the point at which the piping is decoupled. Mass effects of the branch line are considered in the analysis of the run line.

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The branch point is considered as an anchor in the analysis of the branch pipe. Thermal and seismic anchor movements and seismic inertia effects of the run pipe on the branch pipe will be considered where significant. On the basis of its review of the decoupling requirements in the CESSAR-DC, the staff concludes that the requirements presented above are adequate to ensure dynamic decoupling of branch lines from the main run, and thus are acceptable.

3.12.4.5 Conclusions

ABB-CE submitted acceptable piping modeling criteria in its CESSAR-DC that reflect the staff's positions stated above. The staff concludes that ABB-CE meets Appendix B to 10 CFR Part 50 and GDC 1 by submitting information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I piping designated as ASME Code Class 1, 2, and 3 and those not covered by the code within the present state-of-the-art limits and by having design control measures that are acceptable for ensuring the quality of its computer programs. Although COL applicants or licensees referencing the System 80+ design are not required to use the ABB-CE computer programs, the staff will require that computer programs used by the COL applicant or licensee to complete its analyses of the System 80+ piping systems be validated using the piping benchmark program discussed herein.

3.12.5 Pipe Stress Analysis Criteria

3.12.5.1 Seismic Input

CESSAR-DC Appendix 3.9A, Section 1.4.3.2.1.2, presents several options of seismic input for piping analysis. The response spectrum curves for System 80+ were developed using three control motion time-history analyses. The analyses covered a wide range of possible soil and foundation conditions. Piping response spectrum analysis may be performed using one of the following four options of seismic input:

- (1) The first option involves broadening of the raw response spectra, initially, by ± 15 percent for all soil cases. The envelope of the broadened spectra of all soil cases is then used directly in the design. This option is used wherever possible. When it is too conservative, one of the other three options may be used.
- (2) Broadening of the raw response spectra by ± 15 percent is performed for all soil cases. Grouping of the sites is then performed according to site categories (e.g., soft sites, medium sites, and hard sites). An envelope of the broadened spectra for each site category is developed and used in the design process.
- (3) Peak broadening or peak shifting of the raw response spectra by ± 15 percent is performed for all soil cases. The resulting spectra for each individual soil case are then used in the design process. The piping design will be analyzed for each soil case individually and the maximum response from all the cases analyzed will be used.
- (4) The final option is to apply the site-specific response spectra. The site specific analyses will be performed using the site soil conditions and properties and the design control motion (CMS1, CMS2, or CMS3 anchored to a 0.3g PGA) which envelopes the site requirements and is appropriate for the site conditions.

The staff found these four seismic input options for response spectrum analysis acceptable. The options provide the needed flexibility to ensure adequate safety margins in piping seismic design. However, in reviewing the sample piping analysis problems during the May 1993 audit, the staff had some concerns about the peak shifting procedure used by ABB-CE. The shift factor was calculated on the basis of the response spectrum peak in one direction. The same factor was then applied in all three directions. The staff noted that the shifting process should be performed independently in each of the three directions. Subsequently, ABB-CE submitted a markup to revise CESSAR-DC Appendix 3.9A, Section 1.4.3.2.1.2 to reflect this position. The staff accepted the proposed revision. The revision was incorporated into the CESSAR-DC by Amendment R.

3.12.5.2 Design Transients

CESSAR-DC Section 3.9.1.1 discusses the design transients for ASME Code Class 1 components and supports. CESSAR-DC Table 3.9-1 lists the design transients for five plant operating conditions and the number of occurrences for each of the design transients that will be used in the design and fatigue analyses of the ASME Code Class 1 piping systems.

The operating conditions are

- ASME Service Level A: normal conditions
- ASME Service Level B: upset conditions - incidents of moderate frequency

- ASME Service Level C: emergency conditions - infrequent incidents
- ASME Service Level D: faulted conditions - low-probability postulated events
- testing conditions

The CESSAR-DC states that the System 80+ events and total number of occurrences conservatively represent the 60-year design basis. The System 80+ design-basis events are nearly identical to System 80. The 60-year frequency of occurrence for all events was recomputed on the basis of the latest industry databases. Where appropriate, excessive conservatism was removed from the System 80 values. Normal and test event frequencies were determined by summing the number of expected plant operations over the 60-year design life. The frequencies for upset, emergency, and faulted events were determined on a probabilistic basis. ABB-CE notes that the stated 60-year design frequency of occurrence given in Table 3.9-1 is always greater than the expected frequency of occurrence. On the basis of its review of the above information, the staff concludes that the number of design transients presented in the CESSAR-DC is acceptable.

3.12.5.3 Loadings and Load Combinations

The staff reviewed the methodology for load combinations and the selected values of allowable stress limits. The design criteria for all ASME Code piping using the load combinations and stress limits are given in CESSAR-DC Section 3.9.3.1 and Appendix 3.9A, Section 1.3. For ASME Code Class 1, 2, and 3 piping, the combinations of design loadings are categorized with respect to five service levels, identified as Design level and Levels A, B, C and D. They are shown in CESSAR-DC Table 3.9-10 for Class 1 piping and CESSAR-DC Table 3.9-11 for Class 2 and 3 piping. The stress limits for Class 1 piping for each of the loading combinations are in accordance with ASME Code, Section III, NB-3600. For Class 2 and 3 piping, the stress limits are in ASME Code, Section III, NC/ND-3600. Dynamic responses of piping loadings are combined by the SRSS method in accordance with the guidelines of NUREG-0484, Revision 1.

Consistent with the Commission-approved staff position (SECY-93-087) for eliminating the OBE from design, which is also discussed in Section 3.1.1 of this report, the OBE was not included in the load combinations. ABB-CE did, however, include the additional SSE load combinations for Eq. 12a (Class 1 piping) and Eq. 10b (Class 2 and 3 piping) given in the NRC letter of September 11, 1992, "Safety Evaluation on the Use of a Single Earth-

quake Design for Systems, Structures and Components in the Advanced Boiling Water Reactor," Docket 52-001.

On the basis of its review of this information, the staff concludes that appropriate combinations of normal, operating transients, and accident loadings are specified to provide a conservative design envelope for the design of piping systems. The load combinations are consistent with the guidelines in SRP Section 3.9.3 and the NRC position on single-earthquake design and are acceptable.

3.12.5.4 Damping Values

Damping values that are used in the dynamic analysis of piping systems are presented in CESSAR-DC Section 3.7.1.3 and Appendix 3.9A, Section 1.4.3.2.1.3. CESSAR-DC Table 3.7-1 presents damping values, for various types of structures expressed in percentage of critical damping. For piping with nominal diameter less than or equal to 12 in., 2 percent damping is used for the SSE. For piping with large nominal diameter, 3 percent damping is used. These damping values are in accordance with RG 1.61 and are acceptable.

As an alternative to the RG 1.61 values, variable damping values in accordance with the requirements and limitations of the ASME Code Case N-411-1 may be used. However, the staff's position is that combination of the two damping criteria is not acceptable. CESSAR-DC Table 3.7-1 also states that when the response spectrum method of analysis is used, damping values may be based on Code Case N-411-1 as limited by RG 1.84. The staff finds the use of Code Case N-411-1 damping values acceptable for System 80+ piping systems subject to the conditions given in RG 1.84. Thus when the ASME Code Case damping is used, it should be used completely and consistently for an entire piping system from anchor point to anchor point. It should be limited to the analysis using the response spectrum method. It should not be used for analyzing the dynamic response of piping systems using the supports designed to dissipate energy by yielding (i.e., the design covered by Code Case N-420).

3.12.5.5 Combination of Modal Responses

CESSAR-DC Section 3.7.3.7 and Appendix 3.9A, Section 1.4.3.2.1.5, state that when the response spectrum method of analysis is used, the seismic response of each mode is calculated and combined with the other modal responses using the methods described in RG 1.92. If the modes are not closely spaced, the modal results are combined by the SRSS method. Closely-spaced modes are combined by one of the following methods: grouping method, 10-percent method, or double-sum method. These mode combination methods apply for all modes up to the

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ZPA frequency. The ZPA frequency is the lowest frequency at which the response returns to approximately the ZPA. The staff finds that these methods are consistent with the applicable guidelines of SRP Section 3.9.2 and are acceptable.

3.12.5.6 High-Frequency Modes

For seismic response spectrum analysis, the number of modes considered is generally chosen to correspond with the range of seismic excitation frequencies up to the frequency corresponding to the ZPA. At higher excitation frequencies, pipe members are considered rigid and the acceleration associated with these rigid modes is usually small. In certain situations, however, the response of high-frequency modes can significantly affect support loads, particularly axial restraints on long runs. To account for this possibility, SRP Section 3.7.2 provides guidelines for including sufficient modes in a dynamic analysis to ensure that an inclusion of additional modes does not result in more than a 10-percent increase in responses. The implementation of this requirement may require the inclusion of additional modes with their natural frequencies above the ZPA frequency. However, the SRSS combination of these high-frequency modes is inaccurate and may be significantly unconservative. Appendix A to SRP Section 3.7.2 provides the staff guidelines for the consideration of high-frequency modes in a seismic analysis to account for these "missing mass" effects.

CESSAR-DC Appendix 3.9A, Section 1.4.3.2.1.4, describes two procedures for considering responses associated with high-frequency modes. The first method involves computing the pseudostatic "missing" inertial forces for the high-frequency modes excited at the ZPA. The resulting force system is applied statically to determine the high-frequency mode response. The results of that analysis are then combined with the lower-frequency mode results using a method at least as conservative as the SRSS method. The second method is to compute modal responses for enough modes to ensure that the inclusion of additional modes does not increase the total response by more than 10 percent. Modes with natural frequencies below the ZPA frequency are combined in accordance with RG 1.92. Higher mode responses are combined algebraically with each other. The absolute value of the combined higher modes is then added directly to the total response from the combined lower modes.

On the basis of its review of these procedures, the staff concludes that they are consistent with the guidelines provided in SRP Section 3.7.2, Appendix A, and are

acceptable. These procedures were incorporated into the CESSAR-DC by Amendment R.

3.12.5.7 Fatigue Evaluation for ASME Code Class 1 Piping

ASME Section III requires fatigue analyses on the cumulative damage for all ASME Code Class 1 piping. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by plant operating transients for a 60-year design life. However, recent test data indicates that the effects of the reactor environment could reduce the fatigue resistance of certain materials. A comparison of the test data with the code requirements indicates that the margins in the ASME Code fatigue design curves might be less than originally intended. At this time, the staff is assessing the potential generic implication of this issue on all operating plants. Depending on the severity of the issue, certain actions might be required to generically address this concern.

During the April 1993 staff audit, ABB-CE discussed its position on this issue with the staff. ABB-CE planned to use the current code fatigue curves to account for environmental effects. The staff requested that ABB-CE submit additional information, to justify its position in the System 80+ CESSAR-DC. On July 16, 1993, ABB-CE submitted a markup to revise CESSAR-DC Section 3.9.1.1 which describes ABB-CE's position on environmental effects on fatigue for ASME Code Class 1 components. In the revision, ABB-CE stated that observations of significant environmental degradation of the cyclic behavior of materials in LWR environments are primarily related to high strain ranges, slow strain rates, high oxygen content of LWR primary water environments, high sulfur content of carbon and low alloy steels, and low flow rate conditions. The absence of any one of these conditions is considered to be sufficient to preclude any significant environmental degradation of the fatigue behavior of materials exposed to typical PWR primary coolant environment. Since System 80+ components are not exposed to high oxygen content environments at elevated temperatures, and no carbon or low alloy steel is directly exposed to the primary coolant, no significant environmental degradation of the cyclic behavior of System 80+ components will occur. In the revision, ABB-CE concluded that the existing fatigue curves (S-N curves) contained in Appendix I to ASME Code, Section III, may be used as the basis for analyzing fatigue of System 80+ components, if and when such analyses are required.

On the basis of its review of this information, the staff finds the ABB-CE position acceptable for System 80+ piping systems. The revision was incorporated into the CESSAR-DC by Amendment R.

3.12.5.8 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

During the early review stage, the staff noted that for all ASME Code, Section III, Class 2 and 3 piping components designed for a 60-year design life and subject to cyclic thermal or dynamic fatigue loadings, ABB-CE should commit to performing a cyclic evaluation similar to those required for Class 1 components in Subsection NB of ASME Code, Section III. The staff was concerned that the current ASME Code, Section III, Class 2 and 3 rules for consideration of cyclic loads, which were based on a 40-year plant life, may be inadequate for piping components designed for a 60-year design life. ABB-CE was also asked to discuss how environmental effects will be considered in the fatigue life of Class 2 and 3 piping components.

During the June 1993 staff audit, ABB-CE provided a markup to revise CESSAR-DC Section 3.9.3.1.3 to state that Class 2 and 3 components will be reviewed for thermal fatigue effects using ASME Code, Section III, NC-3219.2, for guidance. Fatigue analysis is performed in accordance with the ASME Code rules in Appendices XIII and XIV for those components that do not meet the NC-3219.2 criteria. Also, in a July 16, 1993 markup, ABB-CE revised CESSAR-DC Section 3.9.1.1 to describe its position on environmental effects on fatigue. This is discussed in Section 3.12.5.7 of this chapter.

On the basis of its review of this information, the staff finds the CESSAR-DC revisions acceptable for System 80+ piping systems. These revisions were incorporated into the CESSAR-DC by Amendment R.

3.12.5.9 Thermal Oscillations in Piping Connected to the RCS

By NRC Bulletin 88-08, the staff requested that licensees and applicants review systems connected to the RCS to determine whether any sections of such piping that cannot be isolated can be subjected to temperature oscillations that could be induced by leaking valves. The original issuance of NRC Bulletin 88-08 was in response to the discovery of cracks in the SI nozzle at Farley, Unit 2. These cracks were attributed to high stresses resulting from thermal stratification in the nozzle. The source of this stratification was determined to be the leakage of colder fluid which passed the check valve, isolating the SI line from the RCS cold leg. Subsequently, Bulletin Supplements 1, 2, and 3 made recommendations for inspection and reported on incidents of apparent outleakage causing line failure at a foreign reactor. These supplements broadened the concern for operating plants to all lines in which stratified flow could occur.

On July 16, 1993, ABB-CE submitted a markup to revise CESSAR-DC Appendix 3.9A, Section 1.4.7, to address this issue. ABB-CE described the program which was established by the owners and operators of ABB-CE plants to respond to the bulletin. The program identified affected piping systems, collected data on thermal effects, and performed stress and fatigue evaluations for several plants. ABB-CE stated that the results obtained from that program are used to evaluate the presence of stratified flow and the influence on thermal stresses in these lines for System 80+.

The results of the program determined that thermal stratification can exist in the isolable portions of the SI lines of certain operating plants. The System 80+ counterpart to the SI line is the DVI line. ABB-CE stated that available data are used to evaluate the thermal stresses in the DVI lines caused by thermal stratification.

Certain plants displayed stratification in the lines between the power-operated relief valves and the pressurizer nozzles. The power-operated relief valves in System 80+ are represented by the SDS. However, ABB-CE stated that the location and orientation of the SDS will minimize the presence of condensed steam believed to be the cause of thermal stratification in these lines. Thus, ABB-CE concluded that thermal stratification should not occur in the System 80+ SDS lines.

The System 80+ SC lines are similar to the arrangements in the CE operating plants: a long section of horizontal line connected through a vertical run of pipe to a nozzle on the bottom of the hot leg. Measurements obtained to date are inconclusive. However, ABB-CE stated that program conclusions will be incorporated in the design of the System 80+ SC lines.

On the basis of its review of this information, the staff concludes that ABB-CE provided an adequate approach and methodology for resolving Bulletin 88-08, including Supplements 1, 2 and 3. The revision was incorporated into the CESSAR-DC by Amendment R.

3.12.5.10 Thermal Stratification

Thermal stratification is a phenomenon that can occur in long runs of horizontal piping when two streams of fluid at different temperatures flow in separate layers without appreciable mixing. Under such stratified flow conditions, the top of the pipe may be at a much higher temperature than the bottom. This thermal gradient produces pipe deflections, support loads, pipe bending stresses, and local stresses. The effects of thermal stratification have been observed in both BWR and PWR feedwater piping as discussed in NRC Information Notice (IN) 84-87 and NRC

IN 91-38. NRC Bulletin 88-11 was issued in response to the results of an inspection of the pressurizer SL at the Trojan plant which showed large, unexpected movements that closed the gaps between the line and pipe whip restraints. The movements were attributed to thermal stratification which might have occurred under certain operating conditions when large temperature differences exist between the RCS and the pressurizer. The bulletin required all PWR licensees to establish and implement a program to assure the structural integrity of the SL when subjected to thermal stratification. The structural reevaluation should consider the cyclic effects of the additional bending stresses in the pipe as well as the local stresses induced by thermal striping (rapid oscillation of the thermal boundary interface along the piping inside surface).

The staff asked ABB-CE to describe how the thermal stratification issue will be considered for the System 80+ piping design in the CESSAR-DC. In its July 16, 1993, submittal to NRC, ABB-CE submitted a markup to revise CESSAR-DC Appendix 3.9A, Section 1.4.7, to discuss its approach for System 80+ piping. ABB-CE described the program sponsored by the owners and operators of ABB-CE operating plants to address Bulletin 88-11. That program obtained data at four operating plants which showed that the maximum temperature differences in the SL walls due to thermal stratification were bounded by the difference in temperature between the pressurizer and the hot leg. ABB-CE stated that the System 80+ SL is designed for the maximum temperature difference that will be experienced between the pressurizer and the hot leg.

CESSAR-DC Appendix 3.9A, Section 1.4.7, states that all piping systems subjected to stratified flow are evaluated for additional thermal stresses due to thermal stratification. Stratification results in local stresses caused by the temperature gradients in the pipe wall and bending stresses caused by the moments generated by the restraining effect of supports on the stratified flow induced curvature of the piping. Two-dimensional, finite-element, heat-transfer and structural thermal stress analyses are performed to determine the rotations and local stresses. A conservative stratified flow thermal hydraulic model with the top half of the fluid at the hot temperature and the lower half at the cold temperature is used to determine the pipe wall temperatures. Bending stresses are determined by allowing the pipe to thermally expand unrestrained and by then applying a set of equal and opposite displacements at the rigid support points. ABB-CE noted that although several experiments and analyses have shown that thermal striping will not significantly contribute to fatigue usage, the striping effects are considered through the use of a one-dimensional (1-D) FEM which determines striping stresses caused by oscillations in the interface region. The striping

stress range is then considered in combination with the stratified flow stress range and combined with other loadings as required to perform the stress and fatigue evaluation in accordance with the ASME Code.

On the basis of its review of this information, the staff finds the analysis methodology to be consistent with that of the work performed to address Bulletin 88-11 in operating reactors, which had been previously accepted by the staff. The staff, therefore, concludes that ABB-CE did submit an acceptable technical approach to adequately address the issue of thermal stratification in the System 80+ piping systems. The revision was incorporated into the CESSAR-DC by Amendment R.

3.12.5.11 Safety-Relief Valve Design, Installation, and Testing

CESSAR-DC Section 3.9.3.3 provides information relating to the design, installation, and testing criteria applicable to the mounting of pressure-relief devices used for the overpressure protection of ASME Code Section III, Class 1, 2, and 3 components. The staff reviewed CESSAR-DC Section 3.9.3.3 in accordance with SRP Section 3.9.3. This review included an evaluation of the applicable loading combinations and stress criteria. The review considered the means offered to accommodate the rapidly applied reaction force when a safety valve or relief valve opens and the transient fluid-induced loads applied to the piping downstream of a safety valve or relief valve in a closed discharge piping system. The method of analysis for safety valves and relief valves accounts for the time history of loads following a valve opening. The fluid-induced forcing functions are calculated for each valve using 1-D equations for the conservation of mass, momentum, and energy. The dynamic response of the piping system is determined by applying these forcing functions at locations of changes in fluid flow direction in a time-history analysis of the piping model. Alternately, an equivalent static analysis may be used following the criteria of Appendix O of the ASME Code Section III as supplemented by the additional criteria of SRP Section 3.9.3, Section II.2.

The staff finds that the information in CESSAR-DC Section 3.9.3.3 meets the applicable guidelines of SRP Section 3.9.3 and is acceptable.

3.12.5.12 Functional Capability

ABB-CE submitted a markup, dated July 16, 1993, to revise CESSAR-DC Section 3.9.3.1.4.3 to state that, to satisfy functional capability requirements, all ASME Code Class 1, 2, and 3 piping systems are designed to meet a pipe D/t ratio of less than 50 in accordance with

NUREG-1367 and the allowable stress from the ASME Code of $3.0 S_m$ (Class 1) or $3.0 S_h$ (Class 2 and 3) but not greater than $2.0 S_y$.

The objective of NUREG-1367 was to determine whether present code rules, and potential changes to the code rules, are sufficient to ensure maintenance of functional capability. Through a review of dynamic testing sponsored by EPRI and the NRC, it was established that stress levels corresponding to the current ASME Code Eq. 9 stress limits for Level D do not result in a loss of piping functional capability provided that certain conditions are met. The conditions are as follows: dynamic loads are reversible in direction, dynamic moments are calculated using an elastic response spectrum analysis with ± 15 percent peak broadening and with no more than 5-percent damping, steady-state stresses do not exceed $0.25 S_y$, D_o/t does not exceed 50, and external pressure does not exceed internal pressure. The staff, therefore, finds the ABB-CE functional capability limits for System 80+ piping acceptable provided that all of the NUREG-1367 conditions are met. The revision was incorporated into the CESSAR-DC by Amendment R.

3.12.5.13 Combination of Inertial and SAM Effects

CESSAR-DC Appendix 3.9A Section, 1.4.3.2.1.6, discusses the methodology for considering seismic anchor movements. The effects of SAM are considered in the seismic analysis by combining them with the SSE inertia effects by the absolute summation. For models with piping in more than one building, the buildings are assumed to move 180 degree out of phase. Movements within all buildings except the RB are assumed to be in phase. Within the RB, there are differential movements between the RB, the containment vessel, reactor interior structures, and the NSSS. These movements are assumed to act 180 degree out of phase. The resulting relative movement is applied as static support displacements with all dynamic supports active.

On the basis of its review of this information, the staff finds the ABB-CE methodology consistent with the applicable guidelines of SRP Section 3.9.2 and, therefore, acceptable.

3.12.5.14 OBE as a Design Load

In SECY-93-087, the staff recommended eliminating the OBE from the design of SSCs in advanced reactors. The Commission approved the staff position in its SRM dated July 21, 1993. In a letter dated September 11, 1992, the staff provided a guidance document which identified the necessary changes to existing seismic design criteria that are acceptable for implementing the proposed rule change

as it pertains to the design of safety-related SSCs for ALWRs. This document described specific supplemental criteria for fatigue, SAM, and piping stress limits which should be applied when the OBE is eliminated. For fatigue evaluation, two SSE events with 10 maximum stress cycles per event (or an equivalent number of fractional cycles) should be considered. The effects of SAMs due to the SSE should be considered in combination with the effects of other normal operation loadings that might occur concurrently. For Class 1 primary stress evaluation, seismic loads need not be evaluated for consideration of Level B service limits for Eq. 9. However, for satisfaction of primary plus secondary stress range limits in Eq. 10, the full SSE stress range or a reduced range corresponding to an equivalent number of fractional cycles shall be included for Level B service limits. These load sets should also be used for evaluating fatigue effects. In addition, the stress due to the larger of the full range of SSE SAM or the resultant range of thermal expansion plus half the SSE anchor motion range, shall not exceed $6.0 S_m$. This was defined as Eq. 12a. For Class 2 and 3 piping, seismic loads are not required for consideration of occasional loads in satisfying the Level B service limits for Eq. 9. SAM stresses are not required for consideration of secondary stresses in Eq. 10. However, stresses due to the combination of range of moments due to thermal expansion and SSE anchor motions shall not exceed $3.0 S_h$. This was defined as Eq. 10b.

ABB-CE designs the System 80+ plant in accordance with the single-earthquake design based on the SSE. CESSAR-DC Section 3.7.3.2 states that seismic Category I subsystems, components, and equipment are designed for a total of two SSE events with 10 maximum stress cycles per event. Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE 344-1987. The staff finds the elimination of OBE to be consistent with SECY-93-087 and, therefore, acceptable. The OBE elimination is also discussed in Section 3.1.1 of this report.

CESSAR-DC Table 3.9-10 lists loading combinations for Class 1 piping. The SSE (or the lower level equivalent) is included in the loading combination for Service Level B primary-plus-secondary stress (Eq. 10) as well as for fatigue evaluation. The SSE anchor movement stresses are also included in Eq. 12a. CESSAR-DC Table 3.9-11 lists loading combinations for Class 2 and 3 piping. SSE anchor movement stresses are included in Eq. 10b. The staff finds these revised stress limits consistent with the Commission-approved staff recommendation for eliminating OBE from piping design and, thus, are acceptable.

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3.12.5.15 Welded Attachments

The requirements for welded attachments are presented in CESSAR-DC Appendix 3.9A, Sections 1.6.5 and 1.7.2.12. CESSAR-DC states that, per the requirements of ASME Code, Section III, Subarticle NC/ND 3645, external and internal attachments to piping are designed so as not to cause flattening of the pipe, excessive localized bending stresses, or harmful thermal gradients in the pipe wall. Such attachments are designed to minimize stress concentrations in applications where the number of stress cycles, due either to pressure or thermal effects, are relatively large for the expected life of the equipment. The local stresses due to all support loads are evaluated and added directly to the nominal pipe stresses at the point of attachment. The combined stresses are compared to the allowable stresses given in CESSAR-DC Tables 3.9-10 and 3.9-11. ABB-CE proposes the use of methods given in Code Cases N-318 and N-392 for evaluating local stresses due to welded attachments.

Code Cases N-318-4 and N-392-1 are acceptable to the staff provided that the conditions given in RG 1.84 are met. CESSAR-DC Table 1.8-7 specifies versions of the code cases as N-318-4 and N-392-1 and the conditions in RG 1.84. RG 1.84 states that applicant should identify in the SAR (1) the method of lug attachment; (2) the piping system involved; and (3) the location in the system where the case is to be applied. This is acceptable. The staff finds that for the System 80+ design certification, this information is not needed to reach a safety conclusion because the information is needed only for documentation purpose and, therefore, is not required for design certification.

3.12.5.16 Modal Damping for Composite Structures

CESSAR-DC Appendix 3.9A, Section 1.4.3.2.1.2, states that composite modal damping is used in piping response spectrum analysis in accordance with the procedure described in CESSAR-DC Section 3.7.2.15. In that section, ABB-CE states that composite modal damping values are used for structures with components of different damping characteristics. The composite modal damping values are based on weighing the damping factors according to the mass or the stiffness of each element. The formulation for mass proportional damping is given in the CESSAR-DC.

The staff discussed the use of composite modal damping in piping systems with ABB-CE during the June 1993 audit. On the basis of the discussions, the staff concludes that it is an acceptable alternate method provided that its application is limited to determining the damping for piping systems that include small-diameter (≤ 30.5 cm (12 in.))

and large diameter (> 30.5 cm (12 in.)) pipes using the RG 1.61 values of 2 percent and 3 percent for the SSE. This information has been incorporated into CESSAR-DC Section 3.7.2.15.

3.12.5.17 Minimum Temperature for Thermal Analyses

CESSAR-DC Appendix 3.9A, Section 1.4.2, discusses the requirements for thermal analysis of System 80+ piping systems. The analysis takes into account forces and moments resulting from thermal expansion and contraction. For all analyses, the ambient temperature is assumed to be 21 °C (70 °F). Flexibility analyses are based on the Young's modulus at the temperature of interest. Stresses are based on the modulus at room temperature as required by the ASME Code. All possible operating modes are evaluated to determine the highest range of thermal expansion stress. The effects of anchor movement due to thermal expansion of equipment or other piping are also considered. However, ABB-CE states that lines with maximum temperature less than 65.6 °C (150 °F) which connect to equipment with thermal movements less than 1.6 mm (1/16 in.) are not analyzed for thermal expansion. The staff reviewed this information and concluded that ABB-CE had defined a reasonable minimum temperature at which an explicit thermal analysis would be performed, because pipe stresses and thermal expansions under 65.6 °C (150 °F) are inconsequential to piping design.

3.12.5.18 Intersystem Loss-of-Coolant Accident (ISLOCA)

In SECY-90-016, dated January 12, 1990, the NRC staff discussed the resolution of the ISLOCA issue for ALWR plants by requiring low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. In its June 26, 1990 SRM, the Commission approved the staff's position as discussed above, provided that all elements of the low-pressure systems are considered. The staff's position is also discussed in Section 3.9.3.1.1 of this report.

By letter LD-93-092, dated June 15, 1993, ABB-CE submitted a report to address the ISLOCA issue. Subsequently, by Amendment Q, this report was incorporated into CESSAR-DC as Appendix 5E. Section 3.3 of CESSAR-DC Appendix 5E states that ISLOCA concerns are addressed by two design options: 1) Option A in which the design pressure rating of equipment or systems is increased to at least 40 percent of the RCS normal operating pressure and 2) Option B in which design features are incorporated to terminate and limit the scope of the ISLOCA event. On the basis of the evaluation in Section 3.9.3.1.1 of this report, the staff found that the

increased design pressure of Option A for low-pressure systems was not completely sufficient to attain a piping 90-percent-survival probability goal. To attain this goal, the staff determined that if, in addition to this increased design pressure of Option A, the wall thickness of carbon steel (SA-106 Grade B) and stainless steel (SA-312 Type 304 and Type 316) piping system is of standard weight in piping of diameter 356 mm (14 in.) and less and is Schedule 40 in piping of diameter 406 mm (16 in.) and greater, the 90-percent probability goal for the piping will be achieved. This determination was discussed with ABB-CE. Subsequently, in CESSAR-DC Appendix 5E, Section 3.3, ABB-CE modified the Option A design approach to also specify these minimum wall thicknesses, thereby assuring attainment of the 90-percent-survival probability goal. This resolves DSER Confirmatory Item 3.12.5-1.

The COL applicant should design the low-pressure piping systems that interface with the RCPB in accordance with the above stated criteria.

3.12.5.19 Conclusions

On the basis of this review, the staff concludes that

- The design transients and resulting load combinations with appropriate specific design and service limits for mechanical components and supports are acceptable and meet the applicable portions of GDC 1, 2, 14, and 15; Appendix B to 10 CFR Part 50; Appendix A to 10 CFR Part 100; and SRP Section 3.9.1.
- ABB-CE meets GDC 2 with regard to ensuring the design adequacy of all seismic Category I piping systems and their supports to withstand earthquakes by meeting the positions of RGs 1.61 and 1.92 or acceptable alternatives and by supplying acceptable seismic analysis procedures and criteria that are consistent with applicable guidelines in SRP Section 3.9.2.
- ABB-CE meets GDC 1, 2, and 3 with regard to the criteria to be used for the design and installation of ASME Code, Class 1, 2, and 3 overpressure-relief devices and GDC 14 and 15 with regard to ensuring that the RCPB design limits for NOP, including anticipated operational occurrences, will not be exceeded.
- ABB-CE meets GDC 2 and 4 with regard to the design and service load combinations and associated stress and deformation limits specified for ASME Code, Class 1, 2, and 3 components by ensuring that these systems and components can accommodate the effects of NOP

as well as such postulated events as LOCAs and the dynamic effects resulting from earthquakes.

3.12.6 Pipe Support Criteria

3.12.6.1 Applicable Codes

The staff reviewed the methodology used in the design of ASME Code Class 1, 2, and 3 component supports as described in CESSAR-DC Sections 3.9.3.1.4 and Appendix 3.9A, Section 1.7. The staff also assessed the design and structural integrity of three types of supports: (1) plate and shell, (2) linear, and (3) component standard types. All ASME Code Class 1, 2, and 3 component supports for the System 80+ standard plant will be constructed in accordance with ASME Code, Section III, Subsection NF. In addition, the CESSAR-DC states that for ASTM A500 Grade B tube steel, the NF requirements are supplemented by the weld requirements of American Welding Society (AWS) D1.1, "Structural Welding Code." The staff finds Subsection NF, as supplemented by AWS D1.1 for tube steel, acceptable for the design of piping supports.

3.12.6.2 Jurisdictional Boundaries

CESSAR-DC Appendix 3.9A, Section 1.7.5, defines the jurisdictional boundaries between pipe supports and interface attachment points, such as structural steel, in accordance with the ASME Code, Section III, Subsection NF, 1989 Edition. The staff's review of the jurisdictional boundaries described in the 1989 Edition finds that they are sufficiently defined to ensure a clear division between the pipe support and the structural steel and are acceptable.

3.12.6.3 Loads and Load Combinations

CESSAR-DC Section 3.9.3.1 and Table 3.9-12 provide the loading combinations for the design of piping supports. The combinations of design loadings are categorized with respect to the ASME Code, Service Levels A, B, C, and D. Pipe support members are designed to meet the requirements defined by ASME Code, Section III, Subsection NF. This is acceptable.

3.12.6.4 Pipe Support Baseplate and Anchor Bolt Design

CESSAR-DC Appendix 3.9A, Section 1.7.4, states that concrete anchor bolts for pipe support base plates are designed to meet the requirements of ACI-349 with several additional requirements and exceptions. The additional requirements are intended to ensure that safety factors acceptable to NRC are met, the design strength of concrete

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for a given expansion anchor or group of anchors is greater than the strength of the anchor steel, the effects of shear-tension interaction are accounted for, and minimum edge distance and bolt spacing are considered in determining expansion anchor capacity. In addition, base plate flexibility is accounted for in the calculation of expansion anchor bolt loads. On the basis of its review of this information and the evaluation in Appendix 3A of this report, the staff finds that the baseplate and anchor bolt designs meet the acceptable criteria stated in IE Bulletin 79-02 and are, therefore, acceptable.

3.12.6.5 Use of Energy Absorbers and Limit Stops

During the June 1993 audit, the staff asked ABB-CE to address in the CESSAR-DC the use of seismic restraints other than snubbers. This would include special engineered supports such as energy absorbers and limit stops. ABB-CE was asked to submit a description of the design and analysis methods and modeling assumptions that would be used in designing a piping system with such supports.

Subsequently, ABB-CE submit provided a markup to revise CESSAR-DC Section 3.9.3.4 to state that energy-absorbing and nonlinear piping restraints may be used on System 80+. If used, a description of the methodology used to analyze and design the piping systems incorporating these elements will be provided for approval. This clarification is acceptable to the staff. The revision was incorporated into the CESSAR-DC by Amendment Q.

3.12.6.6 Use of Snubbers

CESSAR-DC Section 3.9.3.4 discusses the use of snubbers as supports for System 80+ safety-related systems and components. It states that for System 80+ snubbers are minimized, to the extent practical, through the use of design optimization procedures. Snubber operability is assured by incorporating analytical, design, installation, in-service, and verification criteria. The elements of snubber operability assurance for System 80+ are consideration of load cycles and travel during normal operation, verification that thermal growth rates do not exceed requirements, accurate characterization of mechanical properties in the analysis, issuance of design specifications and verification that design requirements are met, verification of proper installation, and establishment of ISI and IST programs. In addition, CESSAR-DC Section 3.9.3.4 states that the COL applicant will provide a listing of all safety-related components which use snubbers, in accordance with SRP Section 3.9.3. This is acceptable.

On the basis of its review of the CESSAR-DC, the staff concludes that the information provided is consistent with applicable portions of SRP Section 3.9.3 and is acceptable.

3.12.6.7 Pipe Support Stiffnesses

CESSAR-DC Appendix 3.9A, Section 1.7.2.8, provides requirements for representing pipe supports in a piping analysis model. It states that supports are modeled in the piping analysis by using the actual support stiffness values or by using rigid stiffness values. When the actual support stiffnesses are used, the flexibility of all support components as well as the effects of the building structure are included in the stiffness value. Typically, rigid stiffness values are used for most supports. Actual values are used for flexible supports such as spring hangers. When a support is assumed to be rigid in the piping analysis, a pipe support deflection limit must be met to ensure that the assumption was reasonable. In a markup to revise CESSAR-DC Appendix 3.9A, Section 1.7.2.8, ABB-CE limited a deflection of 1.6 mm (1/16 in.) in the restrained direction based on the greater of the SSE load or the minimum design load given in CESSAR-DC Appendix 3.9A, Section 1.7.2.13. In addition, the maximum deflection was limited to 3.2 mm (1/8 in.) based on the maximum load combination. The staff reviewed this information and found it acceptable. The revisions were incorporated into the CESSAR-DC by Amendment R.

3.12.6.8 Seismic Self-Weight Excitation

CESSAR-DC Appendix 3.9A, Section 1.7.2.3, provides the design requirements for consideration of pipe support seismic loads. It states that seismic inertia and anchor movement forces from the piping are applied to supports that restrain the seismic movement of the piping system. The response of the support itself due to seismic acceleration is also evaluated. Typically, the inertia response of the support mass is evaluated using the response spectrum analysis method with the damping values for welded and bolted structures given in CESSAR-DC Table 3.7-1.

On the basis of its review of the information in the CESSAR-DC, the staff concludes that seismic self-weight excitation in pipe supports is adequately addressed due to conformance with good engineering practice and staff's position.

3.12.6.9 Design of Supplementary Steel

CESSAR-DC Appendix 3.9A, Section 1.7.4, provides the design criteria for the design of pipe supports using supplementary steel. Supplementary steel for pipe supports are designed in accordance with ASME Section III, Subsection NF. The use of Subsection NF is standard industry practice and has been proven to provide adequate design guidelines for the design of structural steel for use as pipe supports. In addition, for ASTM A500 Grade B tube steel members, the NF requirements are supplemented

by the weld requirements of AWS D1.1, "Structural Welding Code." The staff find this acceptable.

3.12.6.10 Consideration of Friction Forces

CESSAR-DC Appendix 3.9A, Section 1.7.2.9, provides the design requirements for the consideration of friction forces on pipe supports. It states that friction forces are considered in the support design under combined dead-weight and thermal loading. The friction forces are applied in both directions of thermal expansion. The friction force is calculated by taking the smaller of CN or KX , where C is the coefficient of friction, N is the component of force normal to the movement, K is the stiffness of the support in the direction of movement, and X is the pipe displacement in the direction of movement. The coefficients of friction are 0.3 for steel on steel and 0.1 for low-friction slide/bearing plates. The staff finds this acceptable.

3.12.6.11 Pipe Support Gaps and Clearances

CESSAR-DC Appendix 3.9A, Section 1.7.2.10, provides the design requirements for support gaps and clearances. It states that small gaps are provided for frame type supports built around the pipe. The gaps allow for radial thermal expansion of the pipe and allow for pipe rotation. Total gaps of 3.2 mm (1/8in.) or less in the restrained direction are negligible and are considered to be zero in the piping analysis. The staff finds this consistent with industry practice and, therefore, acceptable.

3.12.6.12 Instrumentation Line Support Criteria

CESSAR-DC Appendix 3.9A, Section 1.10, provides the design requirements for instrumentation line tubing and supports. In a markup to revise this section, which was submitted to NRC on July 16, 1993, ABB-CE stated that the same design, analysis, and loading considerations that are used for piping and supports are used for tubing. Because of the large amount of safety-related tubing in the plant, bounding analyses are performed. Two support mechanisms are used, free tube spans and tube track supports. The same criteria are used for both support types. Additional support and mounting considerations are the following: tubing routed in two or more structures are verified to have sufficient flexibility to allow for differential displacements, span lengths are chosen and supports and tube details are designed to accommodate heat tracing or insulation requirements or both, all reservoirs, valves, and other in-line components are independently supported, and movement of the root valve between the pipe and the tubing due to seismic and thermal-induced anchor motions are considered.

The staff reviewed this information and found it acceptable due to conformance with good engineering practice. The revision was incorporated into the CESSAR-DC by Amendment R.

3.12.6.13 Pipe Deflection Limits

CESSAR-DC Appendix 3.9A, Section 1.7.4, includes specific design criteria to ensure that the maximum deflections of the piping at support locations for static and dynamic loadings are within allowable limits. It further ensures that the piping design is consistent with the manufacturers' requirements for pipe deflection limits at pipe supports, such as requirements for travel in snubbers and hangers, or with industry practice, such as requirements for the sway angle in rod or strut supports. Maximum deflections are calculated as part of piping analysis results. It is common industry practice to ensure that all deflections are within design limits and cause no unacceptable effects to other safety-related components in adjacent area. The staff finds that this information is consistent with industry practice and, therefore, acceptable.

3.12.6.14 Conclusions

The CESSAR-DC provides acceptable information to address these issues and positions. ABB-CE meets 10 CFR 50.55a and GDC 1, 2, and 4 with regard to the design and service load combinations and associated stress and deformation limits specified for ASME Code, Class 1, 2, and 3 component supports by ensuring (1) that component supports important to safety will be designed to quality standards commensurate with their importance to safety and (2) that these supports will accommodate the effects of NOP as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE.

3.12.7 Criteria for Postulating High-Energy Line Breaks

By GDC 4, the staff requires that SSCs important to safety be designed to be compatible with and to accommodate the effects of the environmental conditions resulting from normal operations, maintenance, testing, and postulated accidents, including LOCAs. It also requires that they be adequately protected against dynamic effects (including the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures and from events and conditions outside the nuclear power plant.

CESSAR-DC Section 3.6.2 describes the design bases for locating breaks and cracks inside and outside containment, the procedure used to define the thrust at the break location, the jet impingement loading criteria, and the dynamic response models and results. On the basis of its

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review of the information in CESSAR-DC Section 3.6.2, as discussed in Section 3.6.2 of this report, the staff finds that the information conforms with the modified guidelines of SRP Section 3.6.2, Revision 2, including BTP MEB 3-1.

By letter LD-92-056 dated April 30, 1992, ABB-CE transmitted a draft of Section 7 of the System 80+ DSDG to the NRC for review. Section 7 of the DSDG gave detailed requirements for the design of piping and details concerning pipe rupture. Subsequently, a complete version of the DSDG, dated January 15, 1993, was submitted for review. Although not part of CESSAR-DC, the DSDG was subjected to a number of staff audits following which information necessary for the staff's safety evaluation was extracted from the DSDG, modified or clarified as necessary, and subsequently added to Appendix 3.9A to the CESSAR-DC by Amendment P.

In DSER Section 3.6.2, the staff noted that in CESSAR-DC Sections 3.6.2.1.3 and 3.6.2.2, ABB-CE documented that dynamic effects of pipe ruptures in the reactor coolant loop, SL, SC line, SI line, and MSL were to be eliminated from the System 80+ plant design basis by LBB evaluations. CESSAR-DC Section 3.6.3 describes the LBB evaluations to be performed to eliminate these dynamic effects, which is discussed in Section 3.6.3 of this report.

By RAI Q252.03 the staff noted that (1) LBB analyses were to be based on specific plant data and (2) pre-approval of LBB procedures was not staff policy. Requirements for LBB analyses were also provided. Subsequently, as a result of interactions (described in part in DSER Section 3.6.3) with the staff, ABB-CE proposed an approach which was generally acceptable to the staff. As documented in DSER Section 3.6.3, ABB-CE proposed to perform bounding LBB analyses based on the acceptance limits of NUREG-1061, Volume 3, to derive acceptable stress limits for the LBB piping which were to be satisfied by COL applicants. The staff asked ABB-CE to submit these bounding LBB analyses for review and approval for design certification. As discussed in Section 3.6.3 of this report, the staff finds these LBB bounding analyses acceptable for the System 80+ design. On the basis of staff's approval of the LBB method for System 80+ design, it is acceptable to eliminate the dynamic effects of the pipe ruptures of those lines which are eliminated by the LBB evaluation and specified in CESSAR-DC Section 3.6.2.1.3.

Coincident with publication of the DSER, a general issue arose concerning the consideration of OBE effects in pipe-break postulation criteria. One of the guidelines in SRP Section 3.6.2, BTP MEB 3-1, states that the analyses for the maximum stresses, stress ranges, and usage factors to

be used for the determination of postulated high- and moderate-energy pipe break and crack locations should be based on loads which include the OBE. In SECY-93-087, the staff recommended the elimination of the OBE in the design process on the basis that it would not result in a significant decrease in the overall plant safety margin. The detailed basis for the staff's recommendation is discussed in Section 3.1.1 of this report. For pipe break postulation without OBE, the staff specifies acceptable deviations from SRP Section 3.6.2. CESSAR-DC Section 3.6.2 incorporates these acceptable deviations for System 80+ design, and, therefore, meets the staff position.

3.12.7.1 High Energy Piping Systems

CESSAR-DC Section 3.6.2.1 provides the criteria to define break and crack locations and configurations in high- and moderate-energy fluid systems both inside and outside containment. The staff reviewed this information in accordance with SRP Section 3.6.2, Revision 2. The review extended to the criteria for ASME Code, Section III, Class 1, 2, and 3, and seismic Category I non-ASME Code high- and moderate-energy system piping and to the criteria for piping in containment penetration areas and in other than containment penetration areas.

The staff finds that the criteria in CESSAR-DC Section 3.6.1.1 for the definition of high- and moderate-energy fluid system piping are consistent with the criteria in SRP Section 3.6.1, BTP ASB 3-1, and are used correctly in CESSAR-DC, Section 3.6.2. High-energy fluid systems inside and outside containment are specified in CESSAR-DC Tables 3.6-3 and 3.6-4, respectively.

3.12.7.2 Pipe Break Criteria Within the Containment Penetration Areas

SRP Section 3.6.2, including BTP MEB 3-1, Revision 2, provides that breaks need not be postulated in portions of high-energy fluid system piping located between the CIVs inside and outside the containment that are designed to meet ASME Code, Section III, Article NE-1120, and additional guidelines specified in BTP MEB 3-1. These guidelines specify an augmented ISI program that must be implemented for those portions of piping within the break exclusion zone.

By RAI Q210.14 and DSER Open Items 3.6.2-3 and 3.6.2-4, the staff asked ABB-CE to revise CESSAR-DC Section 3.6.2.1.4.1, Item F, (Amendment I) criteria for piping in containment penetration areas. The staff also requested ABB-CE to clarify the design and ISI requirements specified in CESSAR-DC Section 3.6.2.1.4.1, Item F.3 (Amendment I), for these break-exclusion zones.

Furthermore, ABB-CE was requested not to revise the break location criteria for ASME Code Section III, Class 1 high- and moderate-energy fluid system piping in areas other than containment penetration areas which were specified in CESSAR-DC Section 3.6.2.1.4.1, Item A, as proposed in response to RAI Q210.14. In response, by Amendment N, ABB-CE modified CESSAR-DC, Section 3.6.2.1.4.1, Item A, and Section 3.6.2.1.4.1, Item F, which had been changed to 3.6.2.1.4.1, Item G, in Amendment N and thereafter, in accordance with BTP MEB 3-1, Revision 2, as requested. On this basis, DSER Open Items 3.6.2-3 and 3.6.2-4 are resolved.

The staff also notes that the OBE is eliminated from the load combination specified in the break postulation criteria in CESSAR-DC Section 3.6.2.1.4.1, Item G.1.(a), for ASME Code Section III, Class 1 piping near CIVs and in Item G.1.(d) for Class 2 piping. On the basis of the OBE elimination discussed in Section 3.1.1 of this report, this is acceptable.

3.12.7.3 Pipe Break Criteria Outside the Containment Penetration Areas

In response to the preceding staff requests in RAI Q210.14 and DSER Open Items 3.6.2-3 and 3.6.2-4, ABB-CE, by Amendment N, adopted the staff position for postulation of pipe breaks for ASME Code Class 1 piping as delineated in SRP Section 3.6.2, BTP MEB 3-1, Revision 2. However, by Amendment P, ABB-CE reinstated the pipe break criteria for the ASME Code Class I piping outside the containment penetration areas as delineated in SRP Section 3.6.2, Revision 1. Before Revision 2 of BTP MEB 3-1 was issued in June 1987, breaks were postulated at intermediate locations between terminal ends of a pipe run if the maximum stress range as calculated by Eq. 10 is greater than $2.4 S_m$ and if the maximum stress range as calculated by either Eqs. 12 or 13 is greater than $2.4 S_m$, where Eqs. 10, 12, and 13 and S_m are as defined in ASME Section III, Subsection NB 3653. This staff position was implemented in many plants operating today. In Revision 2 of BTP MEB 3-1, the same criteria were maintained for break exclusion in the containment penetration areas. However, for other areas, the criteria were revised to require that breaks be postulated at any intermediate locations when only Eq. 10 predicts stresses greater than $2.4 S_m$. The use of Eqs. 12 and 13 was eliminated. This resulted in an inconsistency in the Revision 2 criteria in that they allow higher limits in the containment penetration areas than in other areas. The break exclusion areas should provide a margin greater than (or at least equal to) the margin for areas outside the break exclusion area. To determine the impact of this inconsistency, the staff obtained several independent analyses for both BWRs and

PWRs which compared the number of postulated pipe breaks resulting from the use of Revisions 1 and 2 criteria. These analyses determined that the Revision 2 criteria will result in a significant increase in the number of postulated breaks, which may be counterproductive in terms of enhancing plant safety. Therefore, the staff has recommended that SRP Section 3.6.2 be revised to reinstate the Revision 1 criteria related to allowing the use of Eqs. 12 and 13 for the postulation of intermediate pipe breaks in ASME Code Class 1 piping systems. On the basis of the above evaluation, the staff concludes that it is acceptable to reinstate the Revision 1 criteria related to allowing the use of Eqs. 12 and 13, as described in CESSAR-DC Section 3.6.2.1.4.1, Item A.

By RAI Q210.15 through RAI Q210.24, RAI Q210.28 through RAI Q210.29 and additional comments from the review of Amendment R to the CESSAR-DC, the staff made a number of requests that ABB-CE revise CESSAR-DC Section 3.6.2 to commit to many of the details of the break and crack location and configuration criteria of BTP MEB 3-1, Revision 2. ABB-CE responded to all of these requests by revising CESSAR-DC Section 3.6.2 in accordance with the staff's comments and BTP MEB 3-1, Revision 2. This is acceptable and resolves DSER Confirmatory Item 3.6.2-1.

Subsequently, ABB-CE submitted the report, "Design and Analysis of System 80+ Distribution Systems" (DADS), Revision 1, for staff review. Appendix L of the report summarizes the results of a sample ASME Code Section III Class 2 and 3 stress analysis which included a postulated pipe break analysis. The feedwater economizer line in the RB was used as the sample model. The report referenced the DSDG and a sample piping analysis specification. A draft of this specification, "System 80+ Design Certification Sample Piping Analysis Specification" (SPAS), dated January 15, 1993, was also submitted for staff review. The information in DSDG Section 7.1.8 and SPAS Section 7, both entitled "Postulated Pipe Breaks," are identical to the information in CESSAR-DC Section 3.6.2.

The details of the methodology for pipe-whip analyses of the System 80+ are included in CESSAR-DC Appendix 3.6A. The staff reviewed this appendix and found it acceptable. The staff concludes that proper implementation of (1) the piping design criteria in CESSAR-DC Appendix 3.9A, together with the piping benchmark program and (2) the pipe-whip analysis guidelines in CESSAR-DC Appendix 3.6A will ensure that piping stress analyses for break-postulation and pipe-whip analyses are performed satisfactorily.

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CESSAR-DC Section 3.6.2.2 describes the analytical methods to define the forcing functions and response models for the dynamic effects of pipe breaks. The staff's review of the CESSAR-DC Section 3.6.2.2.1 methodology for determining pipe thrust and jet loads finds that the methods are in accordance with SRP Section 3.6.2, Revision 2, and, hence, are acceptable.

CESSAR-DC Appendix 3.6A provides guidelines for the design and selection of pipe-whip restraints and pipe-rupture and jet-impingement evaluation methods. The pipe-rupture methods include energy balance and dynamic analysis for simplified and detailed models. These guidelines are in accordance with SRP Section 3.6.2, Revision 2, Item III, requirements and, hence, are acceptable. This resolves DSER Confirmatory Item 3.12.7.3-2.

Additionally, the staff conducted an audit review to further assess ABB-CE's dynamic analysis methods for pipe whip. During the audit, the staff reviewed and found acceptable ABB-CE analyses and calculations which demonstrate (1) the determination of the pipe rupture loads in the MSVH to be considered in the design of the MSVH walls and (2) the compliance with the BTP MEB 3-1 B.1.c.(4) criterion that structures separating high-energy lines from essential components be designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect at the structure irrespective of the fact that the break postulation stress criteria might not require such a break location be postulated.

CESSAR-DC Section 3.6.2.3 describes the dynamic analysis methods used to verify the design adequacy of systems, components, and component supports under pipe whip or jet-impingement loadings. By RAIs Q210.31 and Q210.32, the staff asked ABB-CE to clarify the allowable stresses and dynamic load factors used in the pipe-whip analyses as described in CESSAR-DC Section 3.6.2.3. ABB-CE responded to these requests by committing to use a 50-percent ultimate uniform strain criterion and a dynamic load factor of 2 for other than energy-absorbing whip restraint structural elements. These criteria are in accordance with SRP Section 3.6.2, Revision 2, guidelines and, hence, are acceptable.

The staff also noted that the OBE is eliminated from the load combinations specified in the break postulation criteria in CESSAR-DC Section 3.6.2.1.4.1, Item A.2.b, for ASME Code, Section III, Class 1 piping outside the containment penetration area and, in Item B.2.c, for Class 2 and 3 and seismically analyzed B31.1 piping. On the basis of the OBE elimination discussed in Section 3.1.1 of this report, this is acceptable.

3.12.7.4 Conclusions on High-Energy Systems

On the basis of its review of CESSAR-DC Section 3.6.2, the staff concludes that the criteria for postulating pipe break and crack locations and the methodology for evaluating the subsequent dynamic effects resulting from these breaks comply with SRP Section 3.6.2 and meet GDC 4 requirements and, therefore, are acceptable for ensuring that the System 80+ plant design is adequately protected against the effects of postulated high-energy line breaks. The staff's conclusion is based on the following.

The pipe-rupture locations will be adequately determined using the staff-approved criteria and guidelines discussed. The design methods for high energy mitigation devices and the measures to deal with the subsequent dynamic effects of pipe whip and jet impingement are sufficiently and adequately defined by ABB-CE to provide adequate assurance that upon completion of the high-energy line break analyses by the COL applicant, the ability of safety-related SSCs to perform their safety functions will not be impaired by the postulated pipe ruptures.

The provisions for protection against the dynamic effects associated with pipe ruptures of the RCPB inside the containment and the resulting discharging fluid provides adequate assurance that design-basis LOCAs will not be aggravated by the sequential failures of safety-related piping and that the performance of the ECCS will not be degraded as a result of these dynamic effects.

The final arrangement of piping and restraints and the final design considerations for high- and moderate-energy fluid systems inside and outside the containment, including the RCPB, shall be the responsibility of the COL applicant to complete and shall use the staff approved high-energy line break criteria and guidelines discussed to ensure that the SSCs important to safety that are in close proximity to the postulated pipe ruptures will be protected. CESSAR-DC Section 3.6 states that the COL applicant will provide final designs of high- and moderate-energy fluid systems. This is acceptable. In using these criteria and guidelines, the staff is assured that the consequences of pipe ruptures will be adequately mitigated so that the reactor can be safely shut down and maintained in a safe-shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside and outside containment.

3.12.8 LBB Evaluation Procedure

In CESSAR-DC Section 3.6.3.9, Amendment E, ABB-CE stated that Class 1 piping with a diameter of 25 cm (10 in.) or greater and MSL piping meet all the criteria for the

application of LBB. LBB analysis is used to eliminate from the structural design basis the dynamic effects of postulated pipe ruptures.

Under the broad-scope revision to GDC 4 (52 FR 41288; October 27, 1987), the NRC allows the use of advanced technology to exclude from structural design consideration the dynamic effects of pipe ruptures in nuclear power plants, provided it is demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design bases for the piping. The demonstration of low probability of pipe rupture utilizes a deterministic fracture mechanics analysis that evaluates the stability of postulated, small, through-wall flaws in piping and the ability to detect leakage through the flaws long before the flaw could grow to unstable sizes and break the pipe. The concept underlying such analyses is referred to as LBB.

3.12.8.1 LBB Acceptance Criteria

GDC 4 states, in part, that "dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

The analyses referred to in GDC 4 should be based on such specific data as piping geometry, materials, and piping loads. The staff must review the LBB analyses for specific piping design before ABB-CE can exclude the dynamic effects from the design basis. Applicants seeking design certification for ALWRs under 10 CFR Part 52 should establish preliminary stress analysis results, provided that upper and lower bound limits are determined, in order to establish assurance that adequate margins are available for leakage, loads, and flaw sizes. These bounding values and preliminary analyses should be verified when as-built and as-procured information becomes available during the COL phase. Verification of the preliminary LBB analysis should be completed at the COL stage on the basis of actual material properties and final, as-built piping analysis as part of ITAAC associated with 10 CFR Part 52 before fuel loading. This staff position on LBB application is given in SECY-93-087 and was approved by the Commission in its SRM dated July 21, 1993.

A margin of 10 on leakage is required so that leakage from the postulated flaw size is assured of detection when the pipe is subjected to NOP loads. A margin of $\sqrt{2}$ on loads is required to ensure that leakage-size flaws are stable at normal plus accident loads (e.g., SSE and safety-relief

valve discharge loads). A factor of 2 between the leakage-size flaw and the critical-size flaw is required to ensure an adequate stability margin for the leakage-size flaw. The analysis must be performed for an entire pipe run from anchor to anchor.

In addition, applicants seeking approval of LBB during the design certification phase for an ALWR, will be required to perform LBB analyses to establish through-wall flaw sizes and flow stability. For through-wall flaw sizes, a lower-bound, normal-operational stress limit must be established for dead weight, pressure, and thermal loadings. The mean or best-estimate stress strain curve should be used. For flaw stability, an upper-bound stress limit should be established for normal loadings plus SSE. A lower-bound stress-strain curve for base metal should be used regardless of whether the weld or base metal is limiting. In addition, a lower-bound toughness (weld metal or base metal) should be used.

A deterministic fracture mechanics evaluation accounting for material toughness is required. Applicants may propose any fracture mechanics evaluation method for NRC staff review. However, ABB-CE will have to validate the method by comparing it to other acceptable methods or to experimental data.

Using such an approach, an initial set of bounding values and a preliminary LBB analysis should be established during the design certification phase. These bounding values and preliminary analyses can be verified when as-built and as-procured information becomes available during the COL phase. Verification of the preliminary LBB analysis should be completed at the COL stage based on actual material properties and final, as-built piping analysis as part of ITAAC associated with 10 CFR Part 52 before fuel loading.

3.12.8.2 LBB Limitations

The staff has established certain limitations on the use of the LBB approach for excluding piping that is likely to be susceptible to failure from various degradation mechanisms during service. A significant portion of the LBB review is the evaluation of the susceptibility of the candidate piping in various degradation mechanisms to demonstrate that the candidate piping is not susceptible to failure from these degradation mechanisms. The NRC staff reviews the operating history and measures to prevent or mitigate these mechanisms.

The LBB approach cannot be applied to piping that are susceptible to failure in service from such effects as water hammer, creep, erosion, corrosion, erosion-corrosion, fatigue, thermal stratification, and environmental condi-

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tions. Such piping is excluded because these degradation mechanisms challenge the assumptions in the LBB acceptance criteria. For example, (1) water hammer may introduce excessive dynamic loads that are not accounted for in the LBB analyses and (2) corrosion and fatigue may introduce flaws whose geometry may not be bounded by the postulated through-wall flaw in the LBB analyses. Adhering to the defense in depth principle, piping susceptible to failure from these potential degradation mechanisms is excluded from LBB applications.

The limitations and acceptance criteria for LBB used by the NRC staff are discussed in NUREG-1061, Volume 3, "Evaluation of Potential for Pipe Breaks, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," dated November 1984.

3.12.8.3 LBB Design Basis

The broad-scope rule introduced an acknowledged inconsistency in the design basis by excluding the dynamic effects of postulated pipe ruptures while retaining non-mechanistic pipe rupture for containments, ECCS and EQ of safety-related electrical and mechanical equipment.

The NRC staff subsequently clarified its intended treatment of the containment, ECCS, and EQ in the context of LBB application in a request for public comments on this issue that was published on April 6, 1988 (53 FR 11311). Effects resulting from postulated pipe breaks can be generally divided into local dynamic effects and global effects. Local dynamic effects of a pipe break are uniquely associated with that of a particular pipe break. These specific effects are not caused by any other source or even by a postulated pipe break at a different location. Examples of local dynamic effects are pipe whip, jet impingement, missiles, local pressurization, pipe break reaction forces, and decompression waves in the intact portions of that piping or communicating piping. Global effects of a pipe break need not be associated with a particular pipe break. Similar effects can be caused by failures from such sources as pump seals, leaking valve packings, flanged connections, bellows, manways, rupture disks, and ruptures of other piping. Examples of global effects are gross pressurizations, temperatures, humidity, flooding, loss of fluid inventory, radiation, and chemical condition.

The application of LBB technology eliminates the local dynamic effects of postulated pipe break from the design basis. However, global effects may still be caused by something other than the postulated pipe break. Since the global effects from the postulated pipe break produce a reasonably conservative design envelope, the NRC staff will continue to require the consideration of global effects

for various aspects of the plant design such as EQ, ECCS, and the containment.

The elimination of local dynamic effects of postulated high-energy pipe breaks from the design basis of ALWRs using advanced fracture mechanics analyses (LBB approach) is permitted in the revised GDC 4 of 10 CFR Part 50, Appendix A.

3.12.8.4 ABB-CE LBB Evaluation Approach

The application of the LBB approach to ALWRs seeking design certification under 10 CFR Part 52 is acceptable when appropriate bounding limits are established during the design certification phase using preliminary analyses results and verified during the COL phase by performing the appropriate ITAAC discussed herein. Bounding analyses, as described above, must be performed by ABB-CE for ALWR piping requesting the application of LBB. NUREG-1061, Volume 3, constitutes the current NRC-approved LBB methodology and acceptance criteria.

By letter LD-92-064 dated May 8, 1992, ABB-CE stated that it will perform bounding LBB evaluations based on preliminary pipe design analyses for each LBB piping using the guidelines of NUREG-1061, Volume 3. For each piping system evaluated for LBB, potential degradation mechanisms, steam hammer and water hammer, and thermal stratification were to be considered, as applicable. In addition, dynamic strain aging of carbon steel, environmental effects on fatigue, and thermal aging of cast stainless steel piping were to be considered in each LBB evaluation as appropriate. Each LBB piping system was to be evaluated from anchor point to anchor point. Leakage detection outside the containment would be considered for the MSL if the anchor-to-anchor portion of the piping evaluated includes piping that can leak outside the containment. A leak detection capability of 0.06 L/sec (1.0 gpm) was to be used with a factor of 10 for calculating the length of a leakage crack. ABB-CE also submitted a reference for benchmarking its LBB calculations.

In a meeting with ABB-CE on April 22-23, 1992 (meeting summary dated May 28, 1992), ABB-CE proposed not to perform the actual routing of candidate piping for LBB analyses. ABB-CE would use the LBB acceptance limits in NUREG-1061, Volume 3, to derive corresponding acceptable stress limits for the LBB piping. ABB-CE stated that the acceptable stress limits would form a "window" in stresses and the COL applicant would have to verify that the piping is within this window during the COL stage to justify the application of LBB. This window concept is a new approach in meeting the staff's established LBB criteria. The staff noted that ABB-CE should base the bounding LBB analyses on the actual pipe routing.

Since ABB-CE does not intend to complete the actual pipe routing, the staff cannot rule out the possibility that the COL applicant will not be able to satisfy this theoretical window. If the COL applicant cannot meet the limits in the window, appropriate hardware will have to be added for protection against a postulated pipe break. The hardware modification would be feasible except for the building structure that may already have been constructed. Thus, the staff noted that if the bounding LBB analyses are not based on actual pipe routing, the building subcompartments should be designed to accommodate a postulated pipe break of the LBB candidate piping.

Subsequently, the staff found that preliminary pipe routings were used in the bounding LBB analyses, and that stresses in LBB candidate piping systems were within the acceptance limits or window based on NUREG-1061, Volume 3. The preliminary routings were of conventional design. The stresses met the ASME Code and the LBB acceptance limits with ample margins. Consequently, the earlier staff concern was alleviated because it is extremely unlikely the actual routings of the LBB candidate piping systems in COL stage will not be able to meet the acceptance window limits. The staff also found that the related subcompartment walls were sized to accommodate pressurization of the postulated pipe breaks. Thus, the staff review concluded that this ABB-CE approach was generally acceptable and, by DSER Open Item 3.6.3-1, asked ABB-CE to submit its bounding LBB analyses for staff review and approval for design certification.

As part of the LBB bounding analysis review, the staff noted in the DSER that it would also evaluate ABB-CE's justification for using a conversion factor of 2.4 L/sec/cm² (250 gpm per in.²) of leak area for estimating leakage and ABB-CE's justification for combining loads for the flaw stability analysis. By letter dated May 8, 1992 (LD-92-064), ABB-CE further clarified these two subjects. Previously, for estimating leakage, the staff accepted the use of state-of-the-art estimation procedures, such as the EPRI "PICEP" computer program referenced in CESSAR-DC Section 3.6.3.3.2. In DSER Section 3.6.3, the staff also stated that (1) the staff was concerned that ABB-CE is not applying state-of-the-art technology in estimating leakage and (2) for stability analysis, the staff was concerned that ABB-CE was not combining the appropriate transient loads for stability analysis. Subsequently by Amendment R to CESSAR-DC, ABB-CE committed in Appendix 3.9A, Section 1.9.6.2, to use the PICEP program to estimate leakage. The staff review of the bounding analyses of the LBB candidate piping systems as documented in Section 3.6.3.5.3 of this report also found that loads were properly combined in these analyses. Accordingly, the earlier staff concern on appropriateness of combining the transient loads was resolved.

By DSER COL Action Item 3.6.3-1, the staff noted that the COL applicant will be required to verify that the actual material properties and final, as-built piping analyses meet the acceptance parameters established in the bounding LBB analyses. CESSAR-DC Sections 3.6 and 3.6.3.8 incorporate this information. This is acceptable. This is included in COL Action Item 3.6-1.

Details of the staff's review of the ABB-CE bounding LBB calculations appear in Section 3.6.3 of this report.

3.12.8.5 Conclusions on LBB

On the basis of this review, the staff finds that the ABB-CE LBB approach complies with the revised GDC 4 and the Commission-approved staff positions on LBB applications as stated in SECY-93-087. Appropriate bounding limits were established using preliminary analysis results, and conformance with such bounding limits will be verified during the COL phase by performing the appropriate ITAAC. Thus, the probability or likelihood of large pipe breaks occurring in the System 80+ MCL, SL, DVI line, SC line, and MSL piping inside the containment is sufficiently low so that dynamic effects associated with postulated pipe breaks need not be included in the design basis.

3.12.9 Certified Piping Design Requirements

In Table 3.1-1 of ITAAC Section 3.1, entitled "Piping DAC," ABB-CE submitted its certified design commitments for the System 80+ piping design. The staff's evaluation of the certified design commitments and ITAAC for the piping design is given in the following sections.

3.12.9.1 As-Built Piping Verification

ABB-CE identified a certified design commitment that the as-built piping will be reconciled with the as-designed piping configurations. The COL applicant will inspect the piping systems to confirm as-designed conditions. The inspection will verify the pipe routing, support and equipment types and locations, required clearances, and orientation of valve operators. The acceptance criteria require the as-built piping to be reconciled with the piping design requirements in the piping design description.

Originally, the staff review of this ITAAC noted that it did not address the analysis needed to reconcile the differences between the as-built and the as-designed piping systems when discrepancies are found during the inspection. In response, ABB-CE modified this ITAAC to include a reconciliation analysis commensurate with the methods proposed in CESSAR-DC and evaluated in this report,

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which will be documented in the as-built piping report as required by the ASME Code. On this basis, the staff concern is resolved.

The staff position on the tolerances to be used for the reconciliation of the as-built installation of piping systems is that EPRI Report NP-5639, "Guidelines for Piping System Reconciliation (NCIG-05, Revision 1)," dated May 1988, shall be complied with. The staff's acceptance of this approach was documented in a letter from G. Arlotto (NRC) to W. Weber (Nuclear Construction Issues Group) dated February 3, 1988. The staff's endorsement of the EPRI report for the System 80+ design is subject to the restriction that (1) the acceptable as-built piping tolerances not be increased beyond those stated in NP-5639 and (2) they be limited to the piping systems analyzed using linear-elastic methods and qualified on the basis of the staff-approved design criteria specified in this report.

On the basis of these commitments for a COL applicant to verify that the installation tolerances are satisfied and that all deviations are reconciled using the staff-approved methods and DAC discussed in this report, the staff concludes that there is reasonable assurance that the System 80+ piping systems will be constructed in accordance with the design documents.

3.12.9.2 Functional Capability

ABB-CE identified a certified design commitment that all piping systems classified as ASME Code Class 1, 2, and 3 shall be designed to maintain dimensional stability and functional integrity under design loadings expected to be experienced during a 60-year design life. An inspection of the ASME design reports for piping systems classified as ASME Code Section III Class 1, 2, or 3 by the COL applicant will be conducted to assure that the functional capability limits have been satisfied. In Section 3.12.5.12 of this report, the staff evaluated the stress limits to ensure the functional capability of safety-related piping systems. In no case will the piping stress exceed the primary stress limits designated for Service Level D in the ASME Code, Section III. The Service Level D limits are $3.0 S_m$ (not to exceed $2.0 S_y$) for ASME Code Class 1 piping and $3.0 S_h$ (not to exceed $2.0 S_y$) for Class 2 and 3 piping.

The staff finds that the limits specified by ABB-CE to ensure functional capability of piping as discussed in Section 3.12.5.12 of this report are acceptable. The use of Service Level D limits (not to exceed $2.0 S_y$) are consistent with the staff recommendations as documented in NUREG-1367, "Functional Capability of Piping Systems," for ensuring the functional capability of piping systems that were based on high-level dynamic tests sponsored by the EPRI and the NRC staff.

On the basis of these commitments for a COL applicant to limit piping stresses in the certified stress report to the DAC discussed above, the staff concludes that there is reasonable assurance that the piping is capable of performing its safety function under all normal operating, transient, and accident conditions.

3.12.9.3 As-Built LBB Verification

ABB-CE identified a certified design commitment that for each piping system qualified for LBB, the as-built piping and materials will be reconciled with the bases for the LBB acceptance criteria. The COL applicant will perform an inspection of the LBB evaluation report for each piping system qualified for LBB to verify that LBB acceptance criteria are met by the as-built piping and piping materials.

On the basis of these commitments as stated in CESSAR-DC Section 3.9A, Subsection 1.9.6.6 for a COL applicant to inspect as-built piping systems qualified for LBB as discussed above, the staff concludes that there is reasonable assurance that the piping systems qualified for LBB will meet the LBB acceptance criteria, including their material properties.

3.12.9.4 High-Energy Line Break Analysis

ABB-CE identified a certified design commitment that will require an analysis to demonstrate that safety-related systems, components, equipment, and structures are protected against the dynamic effects associated with the postulated break of high-energy piping systems except for those systems to which LBB evaluations have been applied. Seismic Category I SSCs that shall be required to be functional during and following an SSE shall be protected against the environmental effects induced by pipe breaks or cracks, including spraying, flooding, pressure, and temperature effects.

As stated in CESSAR-DC Section 3.6.2, the COL applicant will prepare a pipe break analysis report to verify that the safety of the plant will not be adversely affected by the dynamic and environmental effects resulting from the postulated pipe breaks and cracks. For those impacted components needed to safely shut down the plant, the ASME Code requirements for faulted plant conditions and operability limits shall be met. Pipe break mitigation devices (e.g., pipe whip restraints and jet impingement shields) will be used to restrain the whipping pipe and deflect the blowdown loads. The ITAACs specify that (1) an inspection will be performed to verify the existence of a pipe break analysis report and (2) the as-built high-energy pipe break mitigation features will be inspected.

The COL applicant will complete a pipe break analysis to demonstrate that safety-related SSCs will be protected against the dynamic effects of a postulated pipe break using the methods described in Sections 3.6.2 and 3.12.7 of this report.

On the basis of these commitments for a COL applicant to inspect and analyze pipe breaks as discussed above, the staff concludes that there is reasonable assurance that the safety-related SSCs in the System 80+ are adequately protected against the dynamic effects of postulated high-energy line breaks, and the seismic Category I components will remain functional during and following an SSE.

3.12.9.5 LBB Analysis

ABB-CE identified a certified design commitment that for each piping system qualified for LBB, the as-built piping and materials will be reconciled with the bases for the LBB acceptance criteria.

The COL applicant should perform an evaluation and prepare an LBB analysis report to verify that the safety of the plant will not be adversely affected by dynamic effects resulting from pipe breaks in high-energy piping systems to which LBB evaluation has been applied. The analyses should demonstrate that the designs of these systems are in compliance with the acceptance criteria in the PEDs of CESSAR-DC Appendix 3.9 A or alternately with the criteria of CESSAR-DC Section 3.6.3 and Appendix 3.9A.

On the basis of this commitment, the staff concludes that there is reasonable assurance that the safety-related SSCs in the System 80+ are adequately protected against the dynamic effects of postulated high-energy line breaks in piping systems to which LBB evaluations have been applied.

3.12.9.6 Fracture Toughness

ABB-CE identified a certified design commitment that for those piping systems made of ferritic material as permitted by the design specification, the material chosen will not be susceptible to brittle fracture under the expected service conditions. The material and fabrication process will be selected to reduce the possibility of cracking during service. Chemical, fabrication, handling, welding, and examination requirements that reduce the potential for cracking shall be employed.

In CESSAR-DC Section 5.3.3.1, fracture toughness tests are to be performed in accordance with the requirements of the ASME Code, Section III.

On the basis of this commitment that materials satisfy the requirements of ASME Code, Section III, the staff concludes that there is reasonable assurance that the material for piping systems will be adequately specified to preclude brittle fracture under pressure loadings for the expected service conditions.

3.12.9.7 Cracking in Stainless Steel Piping

ABB-CE identified a certified design commitment that the fabrication process for piping systems made of austenitic stainless steel will be selected to minimize the possibility of cracking during their 60-year design life. Special chemical, fabrication, handling, welding, and examination requirements will be satisfied to minimize the potential for cracking.

On the basis of this commitment, the staff concludes that there is reasonable assurance that the austenitic stainless steel piping systems will be adequately fabricated to minimize the potential for cracking during service.

3.12.9.8 Erosion-Corrosion

ABB-CE identified a certified design commitment that piping systems will be designed to minimize the effects of erosion-corrosion. Erosion-corrosion will be controlled as discussed in CESSAR-DC Section 3.6.3.1.2.

Based on the above certified design commitment for a COL applicant to evaluate the erosion-corrosion effects in piping system, the staff concludes that there is reasonable assurance that the effects of erosion-corrosion are minimized in the piping design for the 60-year plant life.

3.12.9.9 Conclusions

On the basis of the discussions in Section 3.12.9 of this report, the staff concludes that the certified design commitments and the ITAACs for piping design incorporate the staff's approved piping design criteria and analysis methods for ensuring that the piping systems are adequately designed to perform their safety-related functions for all postulated combinations of normal operating, operating transient, and accident conditions, and give reasonable assurance that the piping systems will be built in conformance with the certified design.

3.12.10 Overall Conclusions

The staff concludes that ABB-CE satisfies 10 CFR 52.47 by giving sufficient information to reach a final safety conclusion on all safety questions associated with the

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System 80+ piping design. The staff further concludes that ABB-CE satisfies the applicable requirements of 10 CFR Parts 50, 52, and 100 by providing reasonable assurance that the piping systems will be designed and built in accordance with the certified design and will perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. The staff's conclusion is based on ABB-CE's commitments to pre-approved piping analysis methodology as a part of the DAC which must be used by the COL applicant to complete the plant-specific design and analyses of the System 80+ piping systems. The implementation of these preapproved methods and satisfaction of the acceptance criteria will be verified through the performance of the ITAACs by the licensee to ensure that the final as-built piping stress analyses and high-energy line break analyses as well as the as-constructed piping systems are in accordance with the certified design commitments.

Any change to the commitments involving the piping analysis methodology discussed in Section 3.12 would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Any requested change to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

Appendix 3A

Staff Position on Steel Embedments

Introduction

Vendors of advanced light-water reactors (ALWRs), in their standard safety analysis reports (SSARs) for facilities adopting such reactors, have proposed the use of the American Concrete Institute (ACI) 349 Code for the design of seismic Category I structures. The staff reviewed Appendix B to ACI 349-85, "Code Requirements for Nuclear Safety Related Concrete Structures," and foreign and domestic test data for anchor bolts. On the basis of that review, the staff has taken exceptions to Appendix B to the ACI 349 Code as detailed below. This proposed position has been developed as an aid for the review of the advanced reactors applications. The staff's primary concerns regarding Appendix B to the ACI 349 Code are discussed below and exceptions to the use of Appendix B are noted.

The staff's primary concern with Appendix B to the ACI 349 Code is the use of a basic assumption of the 45-degree concrete-failure cone. This assumption may have been chosen for convenience. However, tests have not confirmed this assumption even for single anchors. The assumption becomes even less conservative when an anchor is located near the free edge of the concrete or when anchors are closely spaced.

Appendix B to the ACI 349 Code is deficient because it has no provisions for reduced anchor strength when the anchor is located in cracked concrete, such as in the tension zone of a concrete slab. The 1988 Edition of the Uniform Building Code has provisions for reduced anchor strength when an anchor is located in the tension zone.

STAFF'S EXCEPTIONS TO Appendix B TO THE ACI 349 CODE

Section B.4.2 - Tension and Figures B.4.1 and B.4.2

In this section and the figures, ACI specifies that the tensile strength of concrete for any anchorage can be calculated by a 45-degree failure-cone theory. The staff disseminated the German test data questioning the validity of the 45-degree failure-cone theory to licensees, architect-engineers, bolt manufacturers, and the code committee members when it met with them. The data showed that the actual failure cone was about 35 degrees and the use of the 45-degree cone theory could be unconservative for anchorages of deep embedment, and for the anchorage of groups of bolts. The code committee, having done some

research of its own, recently agreed with the staff's finding. The code committee is now revising this section. In the meantime, the staff position on issues related to this section is to ensure adoption of design approaches consistent with the test data through case-by-case review.

Section B.5.1.1 - Tension

In this section, ACI states a criterion for ductile anchors. The criterion is that the design pullout strength (force) of the concrete as determined in Section B.4.2 exceeds the minimum specified tensile strength (force) of the steel anchor. Any anchor that meets this criterion is qualified as a ductile anchor and, thus, a low safety factor can be used. The staff believes that the criterion is deficient in two areas: (1) the design pullout strength of the concrete, so calculated, is usually higher than the actual strength, as has been stated in Section B.4.2 above and (2) anchor steel characteristics are not taken into consideration. For example, Drillco Maxi-Bolt Devices, Ltd. claims that its anchors are ductile anchors and, thus, can use a low safety factor. The strength of the Maxi-Bolt is based on the yield strength of the anchor steel, which is 724 MPa (105 ksi). The embedment length of the anchor, which is used to determine the pullout strength of the concrete, is based on the minimum specified tensile strength of the anchor steel of 862 MPa (125 ksi). The staff believes that the 19 percent margin (125/105) for the embedment length calculation is insufficient considering the variability of parameters affecting the concrete cone strength. The staff also questions the energy absorption capability (deformation capability after yield) of such a high-strength anchor steel. Therefore, in addition to the position taken with regard to Section B.4.2 above, the staff will review vendor- or manufacturer-specific anchor bolt behaviors to determine the acceptable design margins between anchor bolt strengths and their corresponding pullout strengths based on concrete cones.

Section B.5.1.1(a) - Lateral Bursting Concrete Strength

This section states that the lateral bursting concrete strength is determined by the 45-degree concrete-failure-cone assumption. This assumption has not been confirmed by tests and the code committee is revising the assumption. The staff believes that the lateral bursting concrete strength determination also needs to be revised. To determine if adequate reinforcement against lateral bursting force needs to be determined on a case-by-case basis, the staff will review the following against test data: (1) the lateral bursting concrete strength provided by the concrete cover around anchor bolts and (2) the lateral bursting force created by the pulling of anchor bolts.

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Section B.5.1.2.1 - Anchor, Studs, or Bars

This section states that the concrete resistance for shear can be determined by a 45-degree half-cone to the concrete free surface from the centerline of the anchor at the shearing surface. Since the assumption for 45-degree concrete-failure cone for tension has not been proved, the staff believes that the use of the 45-degree half-cone for shear should be reexamined. In the meantime, the staff will review the adequacy of shear capacity calculations of concrete cones on a case-by-case basis with emphasis on methodology verification through vendor-specific test data.

Section B.5.1.2.2(c) - Shear Lugs

In this section, ACI states that the concrete resistance for each shear lug in the direction of a free edge shall be determined based on the 45-degree half-cone assumption to the concrete free surface from the bearing edge of the shear lug. This is the same assumption as used in Section B.5.1.2.1, and the staff has the same comment as stated there. Therefore, the staff position related to the design of shear lugs is to do case-by-case reviews. The staff review will emphasize methodology verification through specific test data.

Section B.7.2 - Alternative Design Requirements for Expansion Anchors

In this section, ACI states that the design strength of expansion anchors shall be 0.33 times the average tension and shear test failure loads, which provides a safety factor of 3 against anchor failure. The staff position on safety factor for design against anchor failure is 4 for wedge anchors and 5 for shell anchors unless a lower safety factor can be supported by vendor-specific test data.

Section B.7.2 - Anchors in Tension Zone of Supporting Concrete

When anchors are located within a tensile zone of supporting concrete, the anchor capacity reduction due to concrete cracking shall be accounted for in the anchor design.

Appendix 3B

Staff Position and Technical Bases on the Use of Standard ANSI/AISC N690-1984 Nuclear Facilities: Steel Safety-Related Structures

The use of ANSI/AISC Standard N690-1984 for the design, fabrication, and erection of safety-related structures for the CE System 80+ is in compliance with the guidelines of SRP Sections 3.8.3 and 3.8.4 and is therefore, acceptable when supplemented by the following staff positions and technical bases.

Staff Positions

- (1) In Section Q1.0.2, the definition of secondary stress should apply to stresses developed by temperature loading only.

- (2) Add the following notes to Section Q1.3.6:

"When any load reduces the effects of other loads, the corresponding coefficient for that load should be taken as 0.9, if it can be demonstrated that the load is always present or occurs simultaneously with other loads. Otherwise, the coefficient for that load should be taken as zero."

"Where the structural effects of differential settlement are present, they should be included with the dead load 'D'."

"For structures or structural components subjected to hydrodynamic loads resulting from LOCA and/or safety relief valve actuation, the consideration of such loads should be as indicated in the Appendix to SRP Section 3.8.1. Any fluid structure interaction associated with these hydrodynamic loads and those from the postulated earthquake(s) should be taken into account."

- (3) The stress limit coefficients (SLCs) for compression in Table Q1.5.7.1 should be as follows:

1.3 instead of 1.5 [stated in footnote (c)] in load combinations 2, 5, and 6.

1.4 instead of 1.6 in load combinations 7, 8, and 9.

1.6 instead of 1.7 in load combination 11.

- (4) Add the following note to Section Q1.5.8:

"For constrained (rotation and/or displacement) members supporting safety-related structures, systems, or

components the stresses under load combinations 9, 10, and 11 should be limited to those allowed in Table Q1.5.7.1 as modified by provision 3 above. Ductility factors of Table Q1.5.8.1 (or provision 5 below) should not be used in these cases."

- (5) For ductility factors ' μ ' in Sections Q1.5.7.2 and Q1.5.8, substitute provisions of Appendix A, II.2 of SRP Section 3.5.3 in lieu of Table Q1.5.8.1.

- (6) In load combination 9 of Section Q2.1, the load factor applied to load P_a should be $1.5/1.1 \approx 1.37$, instead of 1.25.

- (7) Sections Q1.24 and Q1.25.10 should be supplemented with the following requirements regarding painting of structural steel:

- Shop painting to be in accordance with Section M3 of LRFD specifications ("Load and Resistance Factor Design (LRFD) Specification for Structural, Steel Buildings and Its Commentary," American Institute of Steel Construction, Chicago, 1986).
- All exposed areas after installation to be field painted (or coated) in accordance with the applicable portion of Section M3 of LRFD specifications.
- The QA requirements for painting (or coating) of structural steel to be in accordance with ANSI N101.4 ("Quality Assurance for Protective Coatings Applied to Nuclear Facilities," American Institute for Chemical Engineers, New York, 1972), as endorsed by RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants."

Technical Bases

- (1) The standard defines the "secondary stress" as: "any normal stress or shear stress developed by the constraint of adjacent material or by self-constraint of the structure. The basic characteristic of a secondary stress is that it is self-limiting due to deformation-limited effects." This definition has been interpreted by some to be applicable to the stresses generated by mechanical (i.e., non-thermal) loads at the structural discontinuities. The position in Section I of this appendix clarifies the staff's interpretation.

- (2) These notes provide guidance to the users regarding consideration of additional load effects in designing steel structures. The notes are parts of SRP Sections 3.8.3 and 3.8.4.

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- (3) The research done in the last 12 years on the strength and stability of compression members shows that the base curve (SSRC curve in Figure 3.1) used in arriving at the SLCs in SRP Sections 3.8.3, and 3.8.4, and in the standard does not reflect the results of the available test data. In developing the American Institute of Steel Construction (AISC) building specification based on the load and resistance factor design (LRFD) concept, the AISC changed the formulation for compression members to reflect the results of the test data. The LRFD curve (with $\phi=1.0$) is also shown in Figure 3.1. Based on the test data, this curve has the minimum reliability index, β (defined as the ratio of $\ln(R_m/Q_m)$ to $(V_R^2 + V_Q^2)^{1/2}$; where R_m = median value of resistance, Q_m = median value of load; and, V_R , and V_Q are the corresponding coefficients of variation), of 2.6 (LRFD Specification for Structural Steel Buildings and Its Commentary, Published by AISC, Chicago, September 1, 1986). The LRFD specification requires $\phi=0.85$ in establishing the resistance of compression members.

Figure 3.1 shows the curves reflecting the SLCs of 1.0, 1.4, 1.5, 1.6, and 1.7 as applied to the stresses specified for allowable stress design of the AISC. Based on the comparison with the LRFD curve ($\phi=1.0$), the following SLCs are recommended in the interim position:

SLC of 1.6 ($\phi \approx 0.95$) for load combinations 10 and 11. This is reasonable for load combinations containing the effects of the two low-probability events, that is, safe-shutdown earthquake (SSE) and loss-of-coolant accident (LOCA).

SLC of 1.4 ($\phi \approx 0.84$) for load combinations 7, 8, and 9. This is appropriate for combinations containing the effects of a single, low-probability event, that is, SSE, tornado, or LOCA.

SLC of 1.3 in load combinations 2, 5, and 6 is recommended when the secondary stresses due to T_o are included in the load combinations. This is consistent with the current position of allowing higher stresses under the effects of operating temperature.

- (4) Neither the standard review plans (SRPs) nor the standard offer any guidance regarding the tolerable deformation of the constrained steel members when they are subjected to temperature growth under sustained T_a or other LOCA loads. Statistically meaningful test data simulating the inelastic behavior of such constrained members under representa-

tive load combinations (including T_a and E_s) are not available. This provision ensures against the instability condition arising from the effects of T_a or other LOCA loads under load combinations 9, 10, and 11.

- (5) The ductility factors provided in Table Q1.5.8.1 are either more liberal than those in Appendix A of SRP Section 3.5.3 (e.g., μ for compression members), or involve some inconsistencies in definitions and interpretation of the formulae (e.g., "formulae" in 2.D of the table) given in the table. Therefore, until sufficient test-based justification for ductility factors listed in Table Q1.5.8.1 is provided, the staff position as stated in the appendix is recommended for use.
- (6) This provision makes the load combination consistent with that in the SRP.
- (7) Additional provision regarding painting of structural steel is provided.

Figure 3.1

STABILITY CURVES FOR AXIALLY LOADED COMPRESSION MEMBERS
 $E = 200 \text{ GPa (29,000 ksi)}$

Appendix 3C

Evaluation of Pumps and Valves Inservice Testing Plan (CESSAR-DC Table 3.9-15)

Introduction

This appendix presents the review results of the ABB-CE System 80+ IST program documented in Section 3.9.6 of the CESSAR-DC and the IST plan, Table 3.9-15 of the CESSAR-DC.

This review was performed to determine if ABB-CE complies with the regulatory and ASME Code, Section XI, 1989 Edition, requirements, and included:

- verification that all pumps and valves classified as safety related in the CESSAR-DC are in the IST plan, as required by Section XI (This includes both passive and active components and the verification that all CIVs contained in Table 6.2.4-1 of the CESSAR-DC are included in the IST plan.)
- a detailed review of the notes that justify for deferral of testing to the cold shutdown or refueling condition
- a review of the testing specified in the plan to ensure that all safety-related functions are tested
- a review of the typical IST connections and configurations (CESSAR-DC Figure 3.9-16)

The review of the IST program was performed using SRP Section 3.9.6; GL 89-04, "Guidance on Developing Acceptable IST Programs;" and the minutes of the public meeting to discuss GL 89-04 on dated October 25, 1989.

Evaluation

The ABB-CE System 80+ pump and valve IST program is documented in Section 3.9.6 of the CESSAR-DC. Specific safety-related pumps and valves subject to IST are listed in Table 3.9-15 of the CESSAR-DC. This table lists the individual pumps and valves and gives the following information for each component in the IST program: identification number, description, safety class, test parameters, test frequency, test configuration and CESSAR-DC figure number. Additionally, the table lists the code category, valve type, actuator, and function. The P&IDs in the CESSAR-DC do not contain all the test connections; therefore, ABB-CE gave typical test configurations to be used for testing and referenced them for each component, as applicable. CESSAR-DC Table 3.9-16 lists

the reactor coolant pressure isolation valves and CESSAR-DC Table 6.2.4 lists the CIVs.

The evaluation of the ABB-CE System 80+ IST program consisted of a detailed review of Table 3.9-15 of the CESSAR-DC and the referenced P&IDs, which are included as figures in the CESSAR-DC. The staff verified that all pumps and valves classified as safety related in the CESSAR-DC are included in the IST plan, as required by the code, and that the testing includes all safety-related functions. Additionally, a review of the P&IDs and the referenced IST connections and configurations (contained in CESSAR-DC Figure 3.9-16) was done to verify that the committed testing in Section 3.9.6 and the table could be completed.

In accordance with OMa-1988 Part 10, as referenced by the 1989 Edition of Section XI, valve testing that is impractical to do during power operation may be deferred to cold shutdown. If testing is also impractical at cold shutdown, it may then be deferred to refueling. Partial-stroke exercising must be done when practical quarterly or at cold shutdowns. ABB-CE submitted numerous notes justifying for deferring testing to cold shutdown or refueling outage, affecting 178 valves. These justifications document the impracticality of testing valves quarterly, during operation, as required by Section XI, and were reviewed to verify their technical bases. As discussed in GL 91-18, it is not the intent of IST to cause unwarranted plant shutdowns or to unnecessarily challenge other safety systems. In general, those tests involving a plant trip, damage to a system or component, or excessive personnel hazards are not considered practical. Removing one train for testing or entering a limiting condition for operation is not a sufficient basis for not performing the required tests, unless the testing renders systems inoperable for extended periods of time. Other factors, such as the effect on plant safety and the difficulty of the test (i.e., the burden) may be considered when determining the impracticality.

Each note justifying for testing during cold shutdown and refueling outage is discussed below.

Notes 1 and 2: Cooling water to RCPs

Exercising the MOVs CC-130, 131, 136, 137, 230, 231, 236, and 237 during operation would isolate cooling water to the RCP support coolers, damaging the pumps. Exercising check valves CC-1507 and 2507, open, and CC-1548 and 2548, closed, requires closing the associated MOVs and would damage the pumps. Therefore, ABB-CE's proposal to test MOVs CC-1507 and 2507, open, and CC-1548 and 2548, closed, and check valves at cold shutdowns when the pumps are not running is acceptable.

Additionally, it is impractical to test check valves CC-1507 and 2507, closed, and CC-1548 and 2548, open, quarterly or at cold shutdowns. These simple check valves have no instrumentation and the only practical means of testing these valves is by leak testing or by the use of test connections. It is impractical to test these valves quarterly or during every cold shutdown because the valves and associated test connections are located inside the containment. Access to testing presents a personnel safety hazard because of high radiation levels and proximity to high-energy systems. Testing these valves during every cold shutdown would be burdensome to the licensee because of the extensive test setup which would require substantial man-hours and the potential for extending the shutdown. Therefore, ABB-CE's proposal to test these valves at refueling outages is acceptable.

Notes 3, 4, and 5: CCW to letdown Hx

To test valves CC-102, 103, 122, and 123 would require isolating the Division 1 nonessential CCW loop, which supplies the normal containment chillers. During NOP, both divisions of chilled water are required to maintain the containment temperature within the TS limit. However, during certain times of the year, only two of the four coolers are required. Therefore, ABB-CE's proposal to test these valves quarterly unless it is impractical because of the heat load, in which case, testing at cold shutdown is acceptable.

Exercising MOVs CC-202, 203, 222, 223, 240, 241, 242, and 243, and check valves CC-2622, open, and 2628, closed, would require isolating the cooling water to the letdown Hx. This would lead to reactor coolant inventory transients that could result in a plant trip and possible damage to the piping due to cavitation. Therefore, testing quarterly is impractical and ABB-CE's proposal to test at cold shutdown is acceptable.

It is also impractical to test valves CC-2622, closed, and 2628, open, at cold shutdown. These simple check valves have no instrumentation, and the only practical means of testing these valves is by leak testing or by the use of test connections. It is impractical to test these valves quarterly or during every cold shutdown because the valves and associated test connections are located inside the containment. Access to testing presents a personnel safety hazard because of high radiation levels and proximity high-energy systems. Testing these valves during every cold shutdown would be burdensome to the licensee because of the extensive test setup which would require substantial man-hours and the potential for extending the shutdown. Therefore, ABB-CE's proposal to test at refueling outages is acceptable.

Notes 6, 7, and 11: Seal injection to RCP

Testing the associated valves during operation would isolate seal injection from the chemical and volume control system to the RCP and result in possible pump damage. Therefore, ABB-CE's proposal to test at cold shutdown when the RCPs are not running is acceptable.

Notes 8, 9, and 10: Charging valves

Testing the associated valves would require isolating the charging and letdown flow, thereby causing reactor coolant inventory transients that could result in a plant trip. Therefore, testing quarterly is impractical and ABB-CE's proposal to test at cold shutdown is acceptable.

Note 12: EFW to steam generators

Testing the associated check valves open would require injecting cold fluid from the EFW storage tank into the steam generators which is impractical to do quarterly because of the potential of damage to the nozzles as a result of thermal shock. Therefore, ABB-CE's proposal to test at cold shutdown is acceptable.

Notes 13 and 14: RCS depressurization and vent valves

These valves are normally closed. The RCS vent valves are used to vent non-condensable gases during postaccident conditions and steam from the pressurizer for RCS pressure control. The RCS rapid depressurization valves do not serve a design-basis safety-related function. They are used to depressurize the RCS for the beyond-design-basis event when normal feedwater and EFW are unavailable to remove core decay heat. These valves require operator action to function. Testing these normally closed valves during operation increases the risk of losing reactor coolant, which would require or cause a plant shutdown and is, therefore, impractical. Therefore exercising these valves at cold shutdown is acceptable.

Notes 15 and 16: Main feedwater and steam isolation valves

The main feedwater and steam valves are open during operation. Testing them quarterly would isolate feedwater and steam to the steam generators, causing a steam generator level transient and possibly a plant trip. Therefore, ABB-CE's proposal to partial stroke the main steam valves quarterly and full stroke both the feedwater and main steam valves at cold shutdowns is acceptable.

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Note 18: SI to the RCS check valves

The check valves referenced in Note 18 open to allow the SI pumps to inject into the RCS. They also perform a safety-related function in the closed direction, some as CIVs or pressure isolation valves (PIVs). Testing quarterly is impractical because the SI pump discharge pressure is lower than the RCS pressure. Full-flow or partial-flow testing at cold shutdown is impractical because of low-temperature overpressurization (LTOP) concerns. Therefore, ABB-CE's proposal to full-flow test at refueling outages is acceptable.

It is also impractical to reverse flow exercise valves SI-113 and 133 quarterly or at cold shutdown. These simple check valves have no instrumentation. The only practical means of reverse flow testing these valves is by leak testing. It is impractical to test these valves quarterly or during every cold shutdown because the valves and associated test connections are located inside the containment. Access to testing presents a personnel safety hazard because of high radiation levels and proximity to high-energy systems. Testing these valves during every cold shutdown would be burdensome to the licensee because of the extensive test setup which would require substantial man-hours and the potential for extending the shutdown. Therefore, ABB-CE's proposal to test these valves at refueling outages is acceptable.

Note 19: SI/SCS to RCS valves

These check valves open to allow the SCS or either the SI or SCS pumps to inject into the RCS. Testing quarterly is impractical because the SI and SCS pump discharge pressure is lower than the RCS pressure. It is also impractical to operate the SI pumps during cold shutdown due to LTOP concerns. Using the SCS pumps, however, the valves can be full-stroke exercised at cold shutdown. Therefore, full-stroke exercising the valves at cold shutdowns is acceptable.

It is also impractical to reverse flow exercise valves SI-123 and 143 quarterly or at cold shutdown. These simple check valves have no instrumentation. The only practical means of reverse flow testing these valves is by leak testing. It is impractical to test these valves quarterly or during every cold shutdown because the valves and associated test connections are located inside the containment. Access to testing presents a personnel safety hazard because of high radiation levels and proximity to high-energy systems. Testing these valves during every cold shutdown would be burdensome to the licensee because of the extensive test setup which would require substantial man-hours and the potential for extending the shutdown.

Therefore, ABB-CE's proposal to test these valves at refueling outages is acceptable.

Note 20: Containment spray check valves

These normally closed check valves open to allow the containment spray pumps to discharge to the spray header. Their safety-related function in the closed direction is to provide containment isolation. The 35.6-cm (14-in.) valves are simple check valves located inside the containment. It is impractical to full-flow test them because the containment would be sprayed down, causing possible equipment damage. Therefore, these valves are supplied with mechanical exercisers. It is impractical to test the valves open with the exercisers or closed, using test connections during power operation or cold shutdown because the valves are located inside the containment and access to testing presents a personnel safety hazard because of high radiation levels. Testing these valves at refueling outages is, therefore, acceptable.

Note 21: SIT discharge valves

The check valves referenced in Note 21 open to allow the safety injection tanks (SITs) to inject into the RCS. Testing quarterly is impractical because the SIT pressure is lower than the RCS pressure. ABB-CE has proposed partial-flow exercising during cold shutdown and full-stroke exercising during refueling outages. In CESSAR-DC, ABB-CE states that full-flow testing during cold shutdown is impractical due to the risk of injecting nitrogen into the RCS and the significant RCS inventory increase. In NRC Information Notice 89-67, the staff discusses a concern with injecting nitrogen into the RCS which can cause the residual heat removal pumps to become gas bound when fuel is in the vessel. This event occurred because of a lack of operator control during testing. Numerous ABB-CE plants, including Palo Verde, do full-stroke testing during refueling outages when the vessel head is removed. ABB-CE's proposal to part-stroke exercise the valves during cold shutdown and full-stroke exercise during refueling outages is acceptable.

Note 22: DVI nozzle check valves

These check valves open to allow the SIT or SI pumps to inject into the reactor vessel and serve as PIVs. Valves SI-227 and 247 also open to allow the SCS pumps to inject into the RPV. Testing quarterly is impractical because the SI and SCS pump discharge and SIT pressure is lower than the RCS pressure. It is also impractical to operate the SI pumps during cold shutdown due to LTOP concerns or

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full-stroke the valves using the SITs due to concerns with nitrogen injection and because the RCS is incapable of accepting the SIT volume when the vessel head is installed. ABB-CE proposes that these valves be partial-stroke exercised during cold shutdown and full-stroke exercised during refueling outages using the SITs. Using the SCS pumps, valves SI-227 and 247 can be full-stroke exercised at cold shutdowns. This proposal is acceptable.

Note 23: SI pump minimum flow valves

Testing the SI pump minimum flow return valve requires both pumps in one division of SI to be inoperable, to prevent operation of the pumps with no flow. Testing is impractical to do quarterly or at cold shutdown because both divisions of SI are required by the TS to be operable (3.5.2 and 3.5.3). During operations, only one train may be inoperable for a limited time (i.e., 72 hours). During shutdown, one division may be inoperable, but only for one hour. Therefore, based on the TS limiting condition for operation (LCO), ABB-CE's proposal to test at cold shutdowns is acceptable.

Note 26: CIVs

These valves are the steam generator blowdown and containment ventilation unit drain header CIVs. These valves function as both CIVs and penetration relief valves and, therefore, have a safety-related function in both the open and closed direction. It is impractical to test the steam generator blowdown valves (xx-032, xx-035) open quarterly because the valves and associated test connections are located inside the containment and testing requires the steam generator blowdown line to be depressurized. Therefore, these valves will be tested at cold shutdown when the steam generators are depressurized. It is impractical to test the containment ventilation drain header valve (xx-162) open quarterly or at cold shutdown because the valve and associated test connections are located inside the containment. Access to testing presents a personnel safety hazard because of high radiation levels and proximity to high-energy systems. Therefore, ABB-CE's proposal to exercise this valve at refueling outages is acceptable.

Note 27: SC purification line CIV

The simple check valve located inside the containment is normally closed. It is only opened during cold shutdown. It is impractical to verify the closure capability of this valve quarterly because the valve is located inside the containment and access to testing presents a personnel safety hazard because of high radiation levels and system

pressure (i.e., RCS pressure). Opening the upstream manual valve outside the containment to allow leakage testing could subject piping to pressures higher than the design, which is also impractical because of the potential for component damage. Therefore, ABB-CE's proposal to test this valve during cold shutdown is acceptable.

Note 28: SIT discharge MOV

These valves are locked open during operation to ensure the capability of SIT injection into the RCS. TS 3.5.1 requires all four SITs to be operable. One tank may be inoperable for 1 hour. Because of the short period of time one tank may be inoperable and the amount of time required to perform the test and restore the tank, testing quarterly is impractical, and ABB-CE's proposal to test at cold shutdown is acceptable.

Note 29: SC suction from RCS

These normally locked-closed valves open to align the SC pump suction to the RCS. Four of the valves are classified as RCS pressure isolation valves. Operating these valves during operation is impractical because the valves are so interlocked that they cannot be opened when the RCS pressure exceeds the SCS operating pressure. Additionally, there would be personnel hazard as a result of leak testing the valve following actuation required by TS SR 3.4.14.1. Therefore, ABB-CE's proposal to test at cold shutdown is acceptable.

Note 30: Containment purge CIVs

These normally closed CIVs are in the high-volume containment purge system. TS 3.6.3 does not allow the 60-cm (24-in.) purge valves to be opened during operation. Testing would require shutting the plant down, which is impractical. Therefore, ABB-CE's proposal to test at cold shutdown is acceptable.

Note 31: Steam generator wet layup CIVs

These check valves are normally closed and are located inside the containment. They are downstream of manual gate valves that are locked closed. The valves are not required to be Appendix J, Type-C tested per CESSAR-DC Table 6.2.4-1. The valves are only open during cold shutdown and this is not a safety-related function. It would be impractical to test these valves quarterly, as the valves are inside the containment and the

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only practical means of testing is by measuring the leakage through a upstream test connection. Therefore, ABB-CE's proposal to test at cold shutdown is acceptable.

Note 32: Normal chilled water to the containment

Testing these normally open valves would require isolating the normal chilled-water system to the containment and ABB-CEDM coolers. During NOP, both divisions of chilled water are required to maintain the containment temperature within the TS limit. However, during certain times of the year, only two of the four coolers are required. Therefore, testing will be done quarterly, unless it is impractical because of the heat load, in which case, testing will then be done during cold shutdown. ABB-CE's proposed testing is acceptable.

Note 33: SIT vent to atmosphere

These locked closed solenoid valves allow the SITs to be vented for refill or to relieve pressure during plant cooldown. Two parallel valves supply isolation between each SIT and atmosphere. The valves have their power removed during operation to prevent inadvertent venting. Testing quarterly would vent the SIT and render it inoperable and, if the tank was not restored within 1 hour, the plant would have to be shut down. Therefore, testing quarterly is impractical, and ABB-CE's proposal to test at cold shutdown, when the SITs are depressurized, is acceptable.

Note 34: Main feedwater check valves

Testing these valves closed during operation is impractical because feedwater flow to the steam generators would have to be isolated causing thermal transients at the nozzles or the plant would have to be shut down. Therefore, ABB-CE's proposal to test the valves at a cold shutdown frequency is acceptable.

Notes 36 and 37: Fire protection water supply and demineralized water CIVs

These normally closed check valves are located inside the containment. The safety-related function of these valves is in the open and closed position. These simple check valves have no instrumentation. It is impractical to test these valves quarterly or during every cold shutdown because the valves and associated test connections are located inside the containment. Access to testing presents a personnel safety hazard because of high radiation levels

and proximity to high-energy systems. Testing these valves during every cold shutdown would be burdensome to the licensee because of the extensive test setup which would require substantial man-hours and the potential for extending the shutdown. Therefore, ABB-CE's proposal to test these valves at refueling outages is acceptable.

Note 39: EFW trip and throttle valves

It is impractical to test the trip and throttle valves quarterly or during cold shutdown due to the required test setup and the length of time to setup, perform the test, and restore the EFW pump. It may not be possible to perform the test within the TS LCO during operation and would be burdensome to do it during cold shutdown as it could extend the outage. These valves will be tested at refueling outages, which is acceptable.

Note 40: IRWST to holdup volume tank (HVT) and reactor cavity flooding MOVs

These normally closed MOVs are opened during severe accidents to flood the reactor cavity to cover core debris. They do not have any design-basis safety-related function. Testing the HVT flooding MOVs during operation or cold shutdown is impractical because manual valves inside the containment must be closed before testing. Testing the reactor cavity flooding MOVs during operation or cold shutdowns compromises the operability of the IRWST. During an accident, water collected in the HVT would be diverted to the reactor cavity, instead of to the IRWST, from which the ECCS pumps take suction. The IRWST is required by the TS to be operable during Modes 1-5. Therefore, ABB-CE's proposal to test these valves during refueling outages is acceptable.

Notes 41 and 42: SI air-operated valves

It is impractical to perform fail-safe testing during power operation or cold shutdown because these valves are located inside the containment. Testing requires the air supply to be locally isolated. Access to testing presents a personnel safety hazard because of high radiation levels and proximity to high-energy systems. Testing these valves during every cold shutdown would be burdensome to the licensee because of the extensive test setup which would require substantial man-hours and the potential for extending the shutdown. Therefore, ABB-CE's proposal to test these valves at refueling outages is acceptable.

Note 44: PIVs

It is impractical to reverse-flow exercise the RCS pressure isolation valves quarterly or during every cold shutdown as these valves are located inside the containment. Access to testing presents a personnel safety hazard because of high radiation levels and proximity to high-energy systems. Testing these valves during every cold shutdown would be burdensome to the licensee because of the extensive test setup which would require substantial man-hours and the potential for extending the shutdown. ABB-CE's proposal to exercise the valves in accordance with TS 3.4.14 is, therefore, acceptable.

Summary and Conclusion

On the basis of the preceding evaluations, ABB-CE has adequately addressed the impracticality of exercising these valves during power operation, and the proposed testing frequencies comply with OMa-1988, Part 10, Sections 4.2.1 and 4.3.2.

ABB-CE's System 80+ IST program complies with the 1989 Edition of Section XI except for the service water and CCW pumps. The code requires pumps to be tested at the same flowrate or pressure (reference value) at every test. The system resistance for the service water and CCW pumps, however, is dependent on the heat load, and it would be impractical to require the test to be done at the same flowrate or pressure. ABB-CE proposes using pump curves in lieu of reference points to verify the operational readiness of these pumps. Pursuant to 10 CFR 50.55a(a)(3)(i), the testing proposed by ABB-CE provides an acceptable level of quality and safety and alternatives to the code, and is, therefore, acceptable.

The ABB-CE System 80+ IST program was reviewed in accordance with the requirements in the 1989 Edition of Section XI and applicable staff guidance as described in Section 3.9.6 of this report. On the basis of the results of this evaluation, the staff concludes that the System 80+ IST Program is in accordance with the commitments in CESSAR-DC Section 3.9.6 and is, therefore, acceptable.

4 REACTOR

4.1 Introduction

Criterion 10 of the general design criteria (GDC) requires that the reactor core and associated systems be designed so that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOO). The staff reviewed the information in the CESSAR-DC in support of the System 80+ standard plant design. The staff's evaluation is presented below.

The System 80+ nuclear steam supply system is designed to operate at a maximum core thermal output of 3,914 megawatts (MWt), with sufficient margin for transient operation and instrument error, without damaging the core and without exceeding the pressure settings of the safety valves in the coolant system.

The reactor is cooled and moderated by light water at a pressure of 1.55×10^4 kPa (2250 psia). The reactor coolant contains soluble boron for neutron absorption. The concentration of the boron is varied, as required, to control relatively slow reactivity changes, including the effects of fuel burnup. Additional boron, in the form of burnable absorber rods, may also be employed to establish the desired initial reactivity. Two other types of burnable absorber rod designs use gadolinium oxide or erbium oxide mixed with uranium dioxide (UO_2). Both full-strength and part-strength control element assemblies (PSCEAs) may be used to compensate for changes in reactivity associated with changes in power level, power distribution, moderator temperature, and boron concentration. The PSCEAs are used primarily for power shape control during power operations. Both types of CEAs are used for reactor shutdown.

4.2 Fuel System Design

The staff based its review of the fuel design on information in CESSAR-DC Chapter 4, "Reactor," and the referenced topical reports. The design was reviewed in accordance with the guidelines provided in the Standard Review Plan (SRP) (NUREG-0800) Section 4.2. The objectives of this fuel system safety review are to assure that (1) the fuel system will not be damaged by normal operation and AOOs, (2) fuel system damage will never be so severe as to prevent control rods from being inserted when they are required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. Not damaged means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements GDC 10, and the design limits that

accomplish this are called SAFDLs. Fuel rod failure means that the fuel rod leaks and that the first fission product barrier, the cladding has, therefore, been breached. Fuel rod failure must be accounted for in the dose analysis required by 10 CFR Part 100 for postulated accidents. Coolability, which is sometimes termed "coolable geometry," is the ability of the fuel assembly to retain its rod bundle geometrical configuration with adequate coolant channel spacing for removal of residual heat. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC (e.g., GDC 27 and 35). Specific coolability requirements for loss-of-coolant accidents (LOCAs) are given in 10 CFR 50.46.

4.2.1 Fuel Assembly Description

The fuel assembly, with 236 fuel rod positions has five guide tubes welded to spacer grids and is closed at the top and bottom by end fittings. The guide tubes each displace four fuel rod positions and serve as channels to guide the CEAs over their entire length of travel.

Incore instrumentation is installed in the central guide tube of selected fuel assemblies.

4.2.2 Fuel Rod Description

The fuel rods consist of slightly enriched UO_2 cylindrical ceramic pellets, a round wire Type 302 stainless steel compression spring and an alumina spacer disc located at each end of the fuel column, all encapsulated within a Zircaloy-4 tube. The fuel rods are internally pressurized with helium during assembly. This internal pressurization, by reducing the stresses from differential pressure, postpones creep collapse beyond the required service life of the fuel and improves the thermal conductivity of the pellet-to-cladding gap.

The fuel cladding is cold-worked and stress-relief-annealed Zircaloy-4 tubing with a wall thickness not less than 0.0584 cm (0.023 in.)

The UO_2 pellets are concave at both ends to better accommodate thermal expansion and fuel swelling. The nominal density of the UO_2 in the pellets is 10.47 g/cc, which corresponds to 95.5 percent of the 10.96 g/cc theoretical density. However, because the pellet dishes and chamfers constitute about 3-percent of the volume of the pellet stack, the average density of the pellet stack is reduced to 10.315 g/cc (nominal). The nominal density of the UO_2 in the pellets may range between 94.5 and 96.5 percent of the theoretical density, with corresponding variations in the nominal pellet stack density.

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The compression spring located at the top of the fuel pellet column maintains the column in its proper position during handling and shipping. The alumina spacer disc at the lower end of the fuel rod reduces the lower end cap temperature, while the upper spacer disc prevents UO_2 chips, if present, from entering the plenum region. The fuel rod plenum, which is located above the pellet column, allows space for axial thermal differential expansion of the fuel column and accommodates the initial helium loading and evolved fission gases.

4.2.3 Burnable Absorber Rod Description

To reduce the beginning-of-life moderator coefficient, fixed burnable neutron absorber rods are included in selected fuel assemblies, replacing or augmenting fuel rods at selected locations. Three alternative burnable absorber rod designs, each mechanically similar to fuel rods, are proposed for the System 80+ design.

The first burnable absorber rod design contains a column of burnable absorber pellets instead of fuel pellets. The absorber material is alumina (Al_2O_3) with uniformly dispersed boron carbide (B_4C) particles. The rest of the column consists of Zircaloy-4 spacers; the total column length is the same as the column length in fuel rods.

The second burnable absorber rod design consists of gadolinium oxide (Gd_2O_3) admixed with natural UO_2 in the central rod portion (axially) and with natural UO_2 at the top and bottom. The total column length is also the same as in fuel rods.

The third burnable absorber rod design consists of erbium oxide (Er_2O_3) admixed with enriched UO_2 in the central rod portion (axially) and with natural, or low enriched UO_2 at the top and bottom. The total column length is also the same as in fuel rods.

Each of these types of burnable absorber rods has been used previously in reactors designed by ABB-CE and has been approved by the NRC. Therefore, all of these designs are acceptable.

4.2.4 Control Element Assembly Description

The control element assemblies (CEAs) consist of either 4 or 12 neutron absorber elements arranged to engage the peripheral guide tubes of fuel assemblies. These elements are connected by a spider structure that couples to the control element drive mechanism (CEDM) drive shaft extension. The neutron absorber elements of a four-element CEA engage the four corner guide tubes in a

single fuel assembly. The 12-element CEAs engage the four corner guide tubes in one fuel assembly and the two nearest corner guide tubes in adjacent fuel assemblies. The CEAs total 93 CEAs, 25 of which are PSCEAs.

The control elements of a four-element full-strength CEA consist of an Inconel 625 tube loaded with a stack of cylindrical silver-indium-cadmium (Ag-In-Cd) absorber bars. The control elements of a 12-element full-strength CEA consist of an Inconel 625 tube loaded with a stack of cylindrical absorber pellets; the absorber material is B_4C pellets of 73-percent theoretical density, except for the lower portion of the elements, which contains reduced-diameter B_4C pellets wrapped in a sleeve of Type 347 stainless steel (felt metal). The control elements of the four-element PSCEA consist of an Inconel 625 tube loaded with Inconel 625 bars over the full active length. Thus, the PSCEAs have lower reactivity worth than the full-strength CEAs and are used primarily for axial power shape control during power operation.

4.2.5 Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters to keep damage to acceptable levels. For convenience, acceptance criteria for these design limits are grouped into three categories in the SRP: (1) fuel system damage criteria, which apply to normal operation and AOOs; (2) fuel rod failure criteria, which apply to normal operation, AOOs, and postulated accidents, and (3) fuel coolability criteria, which apply to postulated accidents.

With one exception, the design bases and the corresponding design limits for the damage mechanisms listed in the SRP, the fuel rod failure thresholds for the failure mechanisms listed in the SRP, and the limits to ensure that coolability is maintained for the severe-damage mechanisms listed in the SRP are presented in the CESSAR-DC and are consistent with the staff's requirements and with those previously accepted in the staff's review of the CESSAR final safety analysis report (FSAR) for the System 80 design. The exception is the proposed use of a statistical convolution technique for determining the number of fuel rods experiencing a departure from nucleate boiling (DNB) during accidents. The DNB statistical convolution approach is discussed in more detail in Section 15.1 of this report.

4.2.6 Design Evaluation

The CESSAR-DC presents methods of demonstrating that the System 80+ design satisfies the established design criteria. These methods include operating experience,

prototype testing, and analytical predictions and are consistent with the staff's requirements and with methods previously accepted in the staff's review of the CESSAR FSAR for the System 80 design.

4.2.7 Evaluation Findings

The staff concludes that the System 80+ design fuel system has been designed so that (1) the fuel system will not be damaged by normal operation and AOOs, (2) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (3) core coolability will always be maintained, even after severe postulated accidents. The fuel system thereby satisfies the related requirements of 10 CFR 50.46, of GDC 10, 27, and 35 of (10 CFR Part 50, Appendix A) 10 CFR Part 50 (Appendix K), and 10 CFR Part 100. This conclusion is based on the following:

- (1) The fuel design has been previously reviewed and approved by the NRC for use in System 80 operating reactors. Surveillance programs to follow the fuel performance of the System 80 design in Arkansas Nuclear One (ANO)-2 and in Palo Verde (Unit 1) show that the fuel is behaving as expected after several cycles of irradiation.
- (2) ABB-CE has submitted sufficient evidence, based on operating experience, prototype testing, and analytical predictions, that these design objectives will be met. The analytical predictions for structural response and CEA ejection were performed in accordance with (a) Regulatory Guides (RGs) 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants" and 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs," and (b) the guidelines for "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" in SRP Section 4.2 (Appendix A).
- (3) ABB-CE has agreed that testing and inspecting new fuel at core loading to ensure conformance with design tolerances is the responsibility of COL applicants and is, therefore, a COL action item. In their facility-specific SAR, COL applicants referencing the System 80+ design certification must discuss on-line fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected. This is COL Action Item 4.2.7-1.
- (4) ABB-CE has described methods of predicting fuel rod failures during postulated accidents so that

radioactivity releases are not underestimated and has thereby satisfied the related requirements of 10 CFR Part 100. The fission product release assumptions used by ABB-CE to meet these requirements are discussed in Section 12.4 of this report.

The staff concludes that the System 80+ fuel system design, including the fuel design, the CEA design, and the initial core design, satisfies all the requirements of the applicable regulations, RGs, and current regulatory positions. The fuel design and the CEA design have been specified and the associated analyses results presented in CESSAR-DC. Startup tests to confirm specified nuclear and thermal-hydraulic design parameters are described in CESSAR-DC Chapter 14.

Any changes to the initial reference design of the fuel, the CEA design, or the initial core design from that presented and evaluated in CESSAR-DC would involve an unreviewed safety question and require prior NRC review and approval prior to implementation, except for the following:

- (1) changes to the design features and evaluated design parameters in Tables 4.1 and 4.2 within the boundaries of the acceptance criteria;
- (2) changes covered by applicable NRC-approved topical reports.

Prior NRC review and approval will also be required to change the NRC-approved analysis methods used to demonstrate conformance of the fuel design, the CEA design, and the initial core design to the design limits in CESSAR-DC. An exception is changes covered by applicable NRC-approved topical reports.

Any requested change to the above information that requires NRC review and approval prior to implementation shall be either specifically described in the COL application or submitted for license amendment after COL issuance. ABB-CE has agreed to these commitments. Therefore, draft safety evaluation report (DSER) Confirmatory Item 4.2.7-1 is resolved.

4.3 Nuclear Design

In evaluating the nuclear design, the staff reviewed the information in the CESSAR-DC, ABB-CE's responses to staff requests for additional information, and the referenced topical reports. The staff followed the guidelines of SRP Section 4.3.

Table 4.1 Design features and acceptance criteria for fuel and initial core design

Design feature	Acceptance criterion*
Fuel design	
Maximum fuel rod burnup	$\leq 60,000$ MWD/MTU
Fuel rod U-235 enrichment	≤ 3.7 wt. %
Er ₂ O ₃ burnable absorber concentration	≤ 2.0 wt. %
Number of integral Er ₂ O ₃ burnable absorber rods per fuel assembly	≤ 124
Natural or low enrichment uranium axial blanket length (top and bottom of fuel and Er ₂ O ₃ burnable absorber rods)	≤ 7.5 inches
Er ₂ O ₃ burnable absorber rod cutback length (top and bottom)	≤ 15.0 inches
Initial core design	
Core power level	≤ 3914 MWt
Cycle length	$\leq 16,000$ MWD/MTU

*Acceptance criteria are nominal values.

4.3.1 Design Basis

ABB-CE presented design bases that comply with the applicable GDC and are, therefore, acceptable. Acceptable fuel design limits are specified (GDC 10), a negative prompt feedback coefficient is specified (GDC 11), and tendency toward divergent operation (power oscillation) is not permitted (GDC 12). ABB-CE design bases require a control and monitoring system (GDC 13) that automatically initiates a rapid reactivity insertion, preventing fuel design limits in normal operation or anticipated transients (GDC 20) from being exceeded. The control system is designed so that a single malfunction or single operator error will not violate fuel design limits (GDC 25). The a reactor coolant boration system is capable of bringing the reactor to cold shutdown conditions (GDC 26), and the control system, combined with the engineered safety

features, can control reactivity changes during accident conditions (GDC 27).

The design will limit the effects of reactivity accidents so that damage to the reactor coolant system (RCS) boundary may not occur (GDC 28).

4.3.2 Design Description

In the CESSAR-DC, ABB-CE describes the first-cycle fuel loading as a three-batch loading scheme in which the type B and C fuel assemblies contain rods of erbia burnable absorber. Fuel enrichment and burnable absorber distributions are shown. Assembly enrichments, core burnup, critical soluble boron concentrations and worths, and plutonium buildup are also presented. Values presented for the delayed neutron fraction and prompt neutron

Table 4.2 Evaluated design parameters and acceptance criteria for fuel and initial core design

Evaluated design parameter	Acceptance criterion*
Core average U-235 enrichment	≤ 2.6 wt. %
Maximum unrodded 3-dimensional peaking factor (F_D)	≤ 2.28
Maximum unrodded integrated radial peaking factor (F_r)	≤ 1.55
Minimum DNBR	≥ 1.24
Net CEA shutdown worth hot full power (HFP)	$\geq 8.86\% \Delta\rho$
MTC (HFP, all rods out)	$\leq -0.1 \text{ E-4 } \Delta\rho/^{\circ}\text{F}$, $\geq -3.5 \text{ E-4 } \Delta\rho/^{\circ}\text{F}$
Power coefficient	$< 0.0 \Delta\rho/(\text{kW}/\text{ft})$
Critical boron concentration	≤ 1056 ppm

*Acceptance criteria are nominal values.

lifetime at beginning and end of cycle are consistent with those normally used and are acceptable.

of uncertainties (SCU) methodology discussed in Section 4.4.

Power Distribution

The design bases affecting power distribution are as follows:

- To meet the initial conditions assumed in the LOCA analysis, the peaking factor in the core will not be greater than 2.28 during normal operation at full power.
- Under normal conditions (including maximum over-power) the peak fuel power will not produce fuel centerline melting.
- During normal operation or AOOs, the power distribution will not cause the departure from nucleate boiling ratio (DNBR) to fall below 1.24, calculated by the CE-1 DNB correlation and the statistical combination

The core operating limit supervisory system (COLSS) continuously monitors important reactor characteristics and establishes margins to operating limits. This system, which consists of software executed on the plant computer, uses the output of the in-core detector system to synthesize the core average axial power distribution. Rod positions taken from the control rod position indication system, together with precalculated radial peaking factors, are used to construct axially dependent, radial power distributions. Using this information, together with measured primary coolant flow, pressure, and temperature, the COLSS establishes the margin to the operating limits on maximum linear heat generation rate and minimum DNBR. The system also monitors azimuthal flux tilt and total power level and generates an alarm if any of these limits are exceeded. The margins to all of these limits except azimuthal tilt are continuously displayed to the operators; the tilt can be displayed at the request of the operator. The operator monitors these margins and takes corrective

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action if the limits are approached. These actions include improving the power distribution by moving full-strength or part-strength rods, reducing power, or changing thermal-hydraulic conditions (i.e., coolant inlet temperature and primary system pressure).

The COLSS algorithms and an uncertainty analysis of the calculations performed by the COLSS are described in Topical Report CENPD-169-P, "COLSS - Assessment of the Accuracy of PWR Generating Limits as determined by the Core Operating Limit Supervisory Systems," which was approved by the staff in NUREG-0852, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System, CESSAR System 80," dated November 1981. Topical Report CEN-356(V)-P-A, Revision 1-P-A, "Modified Statistical Combination of Uncertainties," modified the methodology of the uncertainty analysis in CENPD-169-P. CEN-356(V)-P-A was approved by the staff in its letter to the Palo Verde Nuclear Generating Station, dated October 21, 1987. Therefore, the staff has found the methods employed by ABB-CE in the COLSS to determine power distributions to be acceptable. The COLSS is currently used at ANO-2, San Onofre (Units 2 and 3), Waterford (Unit 3), and Palo Verde (Units 1, 2, and 3).

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on core reactivity of changes in such core conditions as power, fuel and moderator temperature, moderator density, and boron concentration. These coefficients vary with fuel burnup and power level. ABB-CE presents calculated values of the coefficients in the CESSAR-DC and has also evaluated the accuracy of these calculations. The staff reviewed the reactivity coefficients used in the transient and accident analyses and concludes that these coefficients conservatively bound the expected values, including uncertainties. Further, moderator and power Doppler coefficients, along with boron worth, are measured as part of the startup physics testing to ensure that actual values are within those used in these analyses. Although applicant-predicted moderator temperature coefficient (MTC) values for the full range of expected operating conditions during the initial cycle are negative, the technical specifications (TS) that were initially proposed allowed a positive MTC below 100-percent rated thermal power, presumably to encompass future reload cycles. However, the Electric Power Research Institute (EPRI) Advanced Light-Water Reactor (ALWR) Evolutionary Plant Utility Requirements Document (URD) states that the fuel cycle design shall have a nonpositive MTC over the entire fuel cycle when the reactor is critical. The staff agrees with this requirement. As a matter of prudence, the staff believes, reactivity increases that raise moderator temperature should be

self-limiting, ensuring stable power operation. Therefore, the staff will upgrade the safety characteristics of advanced reactors to require that the MTC be nonpositive under all operating conditions. This was designated as DSER Open Item 4.3.2-1. In accordance with this staff requirement, ABB-CE has revised the MTC TS to ensure a nonpositive value under all operating conditions (CESSAR-DC Chapter 16). Therefore, DSER Open Item 4.3.2-1 is resolved.

Control

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is built into the core. ABB-CE has given enough information about core reactivity balance for the first core and has shown that means are incorporated into the design to control excess reactivity at all times.

Both excess reactivity and power level are controlled by moving CEAs and by varying boron concentration in the reactor coolant. In addition, during refueling the chemical and volume control system (CVCS) is capable of shutting down the reactor by adding soluble boron absorber and maintaining the reactor at least 5-percent subcritical. The combination of control systems satisfies the requirement of GDC 26.

System 80+ plants will operate at steady-state full power with one bank of the full-strength CEAs slightly inserted. Limited insertion of the full-strength CEAs permits operators to compensate for fast reactivity changes (e.g., power level changes and the effects of minor variations in moderator temperature and boron concentration) without impairing shutdown capability.

The PSCEAs assist primarily in controlling core power distribution, including xenon-induced axial power oscillations during power operations, and axial power shape during load-following transients. The part-strength CEAs can also control reactivity to compensate for minor variations in moderator temperature and boron concentration during power operations and assist to compensate for reactivity changes due to power level and xenon during load-following transients. The total reactivity worth of the PSCEAs will enable licensees to control load-following transients without changing boron concentrations as much or even at all.

Rod insertion is controlled by the power-dependent insertion limits given in the TS. These limits ensure (1) that sufficient negative reactivity is available to quickly shut down the reactor with ample margin and (2) that, if a control rod were ejected (an unlikely event), the worth

of a control rod would be no worse than the worth assumed in the accident analysis.

Soluble boron absorber is used to compensate for slow reactivity changes, including changes associated with fuel burnup, changes in xenon and samarium concentration, buildup of long-life fission products, burnable absorber rod depletion, and the large moderator temperature change from cold shutdown to hot standby. The soluble boron absorber system can take the reactor at least 10-percent subcritical in the cold shutdown condition.

ABB-CE based the uncertainties in the calculated CEA worths on comparisons of CEA worths with the measured control rod worths from many reactor-years of startup test data for ABB-CE-designed reactors and critical experiments. Based on this, the staff concludes that ABB-CE's assessment of reactivity control is suitably conservative and that the control system has adequate negative reactivity worth to ensure shutdown capability if the most reactive CEA is assumed stuck in the fully withdrawn position. Therefore, the CEA and soluble boron worths are acceptable for use in the accident analysis.

Stability

In the CESSAR-DC, ABB-CE discusses the stability of the reactor to xenon-induced power distribution oscillations and the control of such transients. Because of the negative power coefficient, the reactor is inherently stable to oscillations in total reactor power.

The core may be unstable to axial xenon oscillations during the first cycle. ABB-CE has given the staff sufficient information to show that axial oscillations will be detected and controlled before any safety limits are reached, thus preventing fuel damage. From its analysis, ABB-CE also concluded that the core will be stable to both radial and azimuthal xenon oscillations throughout core life. This conclusion is verified by measurements on an operating reactor, Maine Yankee, in which the predicted damping factor agreed well with the measured value. The staff concurs with this conclusion.

Vessel Irradiation

Maximum fast neutron fluxes greater than 1 million electron volts (MeV) at the vessel inside diameter are presented. For reactor operation at the full power rating and an 80 percent capacity factor, the calculated vessel fluence greater than 1 MeV at the inner vessel wall does not exceed 6.2×10^{19} n/cm² over the 60-year design life of the vessel. ABB-CE had to show that the calculated

fluence complies with the requirements of 10 CFR 50.61. This was designated as DSER Open Item 4.3.2-2. The calculated exposure includes a 30-percent uncertainty factor. The staff reviewed ABB-CE's calculational method and found that calculated values of RT_{PTS} for the maximum value of the fluence and the appropriate base metal and weld chemistry comply with the requirements of 10 CFR 50.61. On this basis, DSER Open Item 4.3.2-2 is resolved.

Criticality of Fuel Assemblies

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and storage facilities. The staff's evaluation is discussed in Section 9.1 of this report.

4.3.3 Analytical Methods

In the CESSAR-DC, ABB-CE describes the computer programs and calculational methods used to calculate the nuclear characteristics of the reactor design. Based on the examples to show that these methods can predict experimental results, the staff concludes that the information presented adequately demonstrates the ability of these analytical methods to calculate the reactor physics characteristics of the System 80+ core.

4.3.4 Summary of Evaluation Findings

To allow for changes of reactivity from reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, ABB-CE has designed a significant amount of excess reactivity into the core. ABB-CE has provided substantial information about core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. ABB-CE has shown that sufficient control rod worth would be available at any time during the cycle to shut down the reactor with at least a 2.0-percent delta-k/k subcritical margin in the hot shutdown condition with the most reactive control rod stuck in the fully withdrawn position.

ABB-CE's assessment of reactivity control requirements over the first core cycle is suitably conservative, and the control system has adequate negative worth to ensure shutdown capability.

ABB-CE describes the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and gives examples to demonstrate the ability of these methods to predict experimental results.

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The information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the CESSAR-DC System 80+ design.

The nuclear design is acceptable and satisfies the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28. This conclusion is based on the following:

- (1) ABB-CE satisfied the requirements of GDC 11 (with respect to prompt inherent nuclear feedback characteristics in the power operating range) by

- (a) calculating a negative Doppler and moderator coefficient of reactivity and
- (b) using calculational methods that have been previously found acceptable (CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," which was approved by the staff in its letter to Combustion Engineering on April 4, 1983).

The staff reviewed the reactivity coefficients in this case and finds them suitably conservative for the reasons stated in Section 4.3.2 of this report.

- (2) ABB-CE satisfied the requirements of GDC 12 (with respect to power oscillations that could result in conditions exceeding SAFDLs) by

- (a) showing that such power oscillations are not possible or can be easily detected and remedied and
- (b) using calculational methods that have been verified by ABB-CE.

The staff has reviewed ABB-CE's analysis of these power oscillations in this case and finds it suitably conservative.

- (3) ABB-CE satisfied the requirements of GDC 13 (with respect to provision of instrumentation and controls to monitor variables and systems that can affect the fission process) by providing

- (a) instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure and
- (b) alarms or control room indications or both for these monitored variables.

- (4) ABB-CE satisfied the requirements of GDC 26 (with respect to provision of two independent reactivity control systems of different designs) by providing a movable control rod system and a liquid absorber system capable of:

- (a) reliably controlling AOOs,
- (b) holding the core subcritical under cold conditions, and
- (c) controlling planned, normal power changes.

- (5) ABB-CE satisfied the requirements of GDC 27 (with respect to reactivity control systems that have a combined capability, in conjunction with absorber addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions) by

- (a) designing a movable control rod system and a liquid absorber system, and
- (b) performing calculations to demonstrate that the core has sufficient shutdown margin with the highest worth rod stuck as discussed in Section 4.3.2 of this report.

- (6) ABB-CE has satisfied the requirements of GDC 28 (with respect to postulated reactivity accidents) by satisfying the following:

- (a) regulatory position in RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,"
- (b) fuel enthalpy limit of 280 cal/gm as shown in Section 15.3.4 of this report, and
- (c) criteria on the capability to cool the core as stated in SRP Section 4.2, and
- (d) using calculational methods that have been found acceptable for reactivity insertion accidents as shown in Table 15.1-3 of this report.

- (7) ABB-CE has satisfied the requirements of GDC 10, 20, and 25 with respect to SAFDLs by demonstrating that

- (a) normal operation, including the effects of AOOs, have met fuel design criteria,
- (b) the automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of AOOs and ensures the automatic operation of systems and components important to safety under accident conditions, and
- (c) no single malfunction of the reactivity control system will violate the fuel design limits.

4.4 Thermal-Hydraulic Design

The scope of the staff's review of the thermal-hydraulic design of the core for the System 80+ design includes the design-basis and steady-state analysis of the core thermal-hydraulic performance. The acceptance criteria used as the bases of the staff's evaluation are given in SRP Section 4.4.

4.4.1 Thermal-Hydraulic Design Bases

The principal thermal-hydraulic design basis for the System 80+ design core is the avoidance of thermal-hydraulic-induced fuel damage during normal steady-state operation and AOOs. To satisfy the design basis, the digital core protection calculator (CPC) will automatically trip or take other corrective action to prevent design limits from being violated. The design analysis was performed and design limits were established on the basis of the criteria in the sections that follow.

4.4.1.1 Departure From Nucleate Boiling (DNB)

The margin to DNB at any point in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The DNBR is defined as the ratio of the heat flux required to produce DNB at the calculated local conditions to the actual local heat flux.

The thermal-hydraulic design basis in CESSAR-DC Section 4.4.1.1 for the DNBR is as follows: "The minimum DNBR shall be such as to provide at least a 95-percent probability with 95-percent confidence (95/95 probability/confidence) that DNB does not occur on a fuel rod having that minimum DNBR during steady-state operation and AOOs."

This design basis is consistent with the guidelines in SRP Section 4.4.II.1 and is acceptable. In CESSAR-DC Section 15.0.4, ABB-CE stated that, for CESSAR-DC

Chapter 15 design-basis events in which the 95/95 DNBR safety limit is violated, the statistical convolution method was used to calculate the number of failed rods. The staff's evaluation of the statistical convolution approach is discussed in Section 15.1 of this report.

4.4.1.2 Hydraulic Stability

The hydraulic stability design basis in CESSAR-DC Section 4.4.1.2 is as follows: "Operating conditions shall not lead to flow instability during steady-state operation or during anticipated operational occurrences." As discussed in Section 4.4.2.3 below, this design basis is acceptable.

4.4.1.3 Core Flow

The minimum allowable reactor coolant flow less a maximum bypass flow (3.0 percent) is the design basis used in the thermal margin analysis. The minimum allowable reactor coolant flow is the total design flow with the four reactor coolant pumps in operation. This defined core flow assures a minimum allowable flow and provides a higher safety margin in the thermal margin analysis; therefore, the staff concludes that the defined core flow is acceptable for use in the thermal margin analysis.

4.4.2 Thermal-Hydraulic Design Methodology

4.4.2.1 Thermal-Hydraulic Analysis Methods

Steady-state thermal-hydraulic analysis for the System 80+ design was performed with the approved thermal-hydraulic code, thermal-hydraulic analytical code (TORC), and the CE-1 critical heat flux correlation. The design thermal-hydraulic margin analysis was performed with the NRC-approved methods: TORC ("TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-A, April 1986) and the fast-running version of the TORC code, CETOP-D ("TORC Code - Verification and Simplified Modeling Methods," CENPD-206-P-A, June 1981; "CETOP-D Code Structure and Modeling Methods for Arkansas Nuclear One Unit 2," CEN-214-A-P, July 1982).

ABB-CE also used the NRC-approved modified statistical combination of uncertainties (MSCU) (i.e., "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A, Rev. 1-P-A, May 1988) for the thermal-hydraulic analysis. With this methodology, the engineering hot channel factors for heat flux, heat input, fuel rod pitch, and cladding diameter are combined statistically with other uncertainty factors to arrive at overall uncertainty penalty factors to be applied to the DNBR calculations performed by the CPCs and the COLSS.

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ABB-CE uses NRC-approved methods for the thermal-hydraulic analysis. This approach is acceptable.

4.4.2.2 Departure From Nucleate Boiling

The correlation used to determine the DNBR is the "CE-1" critical heat flux correlation ("Critical Heat Flux Correlations for CE Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," CENPD-162-A-P, September 1976; "Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Power Distribution," CEN-207-A-P, June 1976). The safety limit DNBR of 1.24 for the System 80+ design gives a 95/95 probability/confidence that the hottest fuel rod will not experience DNB.

The 1.24 value incorporates all applicable penalties, such as for rod bow, the 0.01 DNBR for HID-1 grids, and the penalties specified in the MSCU. The rod bow value used in the analysis is 1.75-percent DNBR for burnups up to 30,000 MWD/MTU. For burnups higher than 30,000 MWD/MTU, because of lower radial power peaks in these higher burnup assemblies and rods, sufficient margins exist to offset the rod bow penalty.

Since the minimum safety limit DNBR of 1.24 was previously approved by the staff (C. Thompson (NRC) letter to W. Conway, APSC, "Issuance of Amendment No. 26 to the Facility Palo Verde Nuclear Generating Station, Unit No. 3," May 20, 1991) for the Palo Verde core design and since the core thermal-hydraulic designs of System 80+ and Palo Verde are similar and the methods used for the thermal-hydraulic analysis are identical, the staff concludes that the minimum design DNBR of 1.24 is acceptable for the System 80+ design calculations. In CESSAR-DC Section 15.0.4 ABB-CE stated that, for CESSAR-DC Chapter 15 design-basis events in which the 95/95 DNBR safety limit is violated, the statistical convolution method was used to calculate the number of failed rods. The staff's evaluation of the statistical convolution approach is discussed in Section 15.1 of this report.

4.4.2.3 Hydraulic Stability

In steady-state, two-phase, heated flow in parallel channels, the potential for hydrodynamic instability exists. ABB-CE gave the following information in the CESSAR-DC to support the contention that the System 80+ core is thermal hydraulically stable. According to the literature cited in CESSAR-DC, flow instabilities that have been observed occur almost exclusively in closed channel systems operating at low pressures relative to pressurized-water reactor (PWR) pressures. For PWR operating pressures, experimental results have shown that, even with

closed channel systems, operating limits caused by critical heat flux are encountered before the flow stability threshold is reached. Kao, Morgan, and Parker (H.S. Kao, C.D. Morgan, and M.B. Parker, "Prediction of Flow Oscillation in Reactor Core Channel," American Nuclear Society Transactions, Vol. 16, p. 212) conducted flow-stability experiments at pressures up to 15.2×10^3 kPa (2200 psia) with closed, parallel, heated channels. They found that, at flow and power levels encountered in power reactors, no flow oscillations could be induced at pressures above 8.3×10^3 kPa (1200 psia). The low resistance to coolant cross-flow among subchannels of fuel assemblies would be expected to have a stabilizing effect, and this expectation is confirmed by experimental results by Veziroglu and Lee (T.N. Veziroglu and S.C. Lee, "Boiling-Flow Instabilities in a Cross-Connected Parallel-Channel or Flow System," American Society of Mechanical Engineers (ASME) Paper 71-HT-12, August 1971), who found that the cross-flow between parallel, heated channels improves flow stability. Experimental evidence that flow instabilities will not occur is given by data from rod bundle DNB tests conducted by ABB-CE (CENPD-162-P-A). Analytical support for the conclusion that flow instabilities will not reduce the thermal margin of ABB-CE's PWRs is contained in a letter to J.F. Stolz, (NRC) from D.H. Williams, Arkansas Power and Light Company, January 16, 1978, and enclosure, "Assessment of Core Flow Stability for Applicant PWRs," CEN-64(A)-P (proprietary) and CEN-64(A) (non-proprietary), July 1977. This document assesses core flow stability for a typical ABB-CE PWR using the CE-HYDNA code (H.B. Currin, et al., "HYDNA-Digital Computer Program for Hydrodynamic Transients in a Pressure Tube Reactor or a Closed Channel Core," report CVNA-77, 1961). It was found that, for nominal coolant conditions, the flow is stable throughout the range of reactor power levels examined (100-percent to 250-percent rated power).

On the basis of past operating experience, flow stability experiments, and inherent thermal-hydraulic characteristics of ABB-CE PWRs, the System 80+ design stability evaluation is adequate.

4.4.3 Loose-Parts Monitoring System

A loose object in the primary coolant system can indicate degraded reactor safety. A loose part may be from a failed or weakened component or may be an item inadvertently left in the primary system during construction, refueling, or maintenance. By frequently hitting other parts in the system, a loose part can contribute to component damage and material wear. A loose part can partially block flow of coolant and cause fuel experiencing DNB to fail. A loose part also increases the likelihood that control

rods will jam and can cause more radioactive crud to accumulate in the primary system. Paragraph 3.5.2.2 in the EPRI URD requires instrumentation to detect the presence of loose parts in the RCS. Early detection of loose metallic parts in the primary system can avoid or mitigate safety-related damage to, or malfunction of, primary system components. As specified in Paragraph II.7 of SRP Section 4.4, the design description and proposed procedures for use of the loose-parts monitoring system (LPMS) should be consistent with the guidance in RG 1.133, Revision 1, "Loose-Part Detection Program for Primary System of Light-Water-Cooled Reactor."

ABB-CE describes the LPMS in CESSAR-DC Section 7.7.1.6.3 and in responses to generic safety issues B-60 and C-12 in CESSAR-DC Chapter 20. The LPMS has two sensors at each natural collection region. The LPMS design complies with RG 1.133, Revision 1. To be consistent with the CE standard TS, the LPMS is not included in the TS for System 80+. However, as ABB-CE states in CESSAR-DC Section 7.7.1.6.3, ABB-CE operating guidelines will call for a channel check at least once per 24 hours, a channel functional check at least once per 31 days, a background noise check at least once 91 days, and a channel calibration at least once per 18 months. These operating guidelines for the surveillance requirements are consistent with the guidelines of RG 1.133, Revision 1. The staff has evaluated the System 80+ LPMS by comparing it with the systems used at other plants, taking into account pertinent differences.

The staff concludes that an acceptable LPMS will be implemented for the System 80+ plants.

4.4.4 Digital Core Protection Calculator and Control Element Assembly Calculator

The CPC and control element assembly calculator (CEAC) system is designed to trip the reactor on low DNBR and or high local power density (LPD), ensuring that the SAFDLs on DNB and centerline fuel melt are not exceeded during A00s. These trips also assist the engineered safety feature systems in limiting the consequences of certain postulated accidents. In CESSAR-DC Section 7.2.1.1.2.5, ABB-CE describes the hardware design of the digital CPC and CEAC. However, ABB-CE did not initially describe the software design, and to complete its review, the staff requested ABB-CE to (1) describe the CPC/CEAC software design and protection algorithm; (2) identify all the differences from the previously approved designs and evaluate how the differences will affect the CPC/CEAC performance and safety functions, and (3) discuss the verification program for implementing the CPC/CEAC software during the certification process. As an alterna-

tive, ABB-CE could have discussed the applicability of the previously submitted and approved documents for the System 80 CPC/CEAC design to the System 80+ design. This was designated as DSER Open Item 4.4.4-1.

In response, ABB-CE revised CESSAR-DC 7.2.1.1.2.5 (Amendment R) to state that the software design of the CPC/CEAC system is described in the following documents:

- "Functional Design Specifications for CPC," CEN-147(S)-P, January 1981
- "Functional Design Requirement for CEAC," CEN-148(S)-P, January 1981
- "CPC/CEAC Software Modifications for Waterford 3," CEN-197(C)-P, March 1982
- "CPC/CEAC Software Modifications for System 80," LD-82-038, March 1982
- "CPC/CEAC Software Modifications for San Onofre Nuclear Generating Station Units Nos. 2 and 3," CEN-281(S)-P, July 1984
- "CPC/CEAC Software Modifications for the CPC Improvement Program," CEN-308-P-A, April 1986
- "CPC/CEAC Software Modifications for the CPC Improvement Program Reload Data Block," CEN-330-P-A, October 1987

and has been reviewed and approved by the NRC in:

- "Safety Evaluation Report Related to Operation of San Onofre Nuclear Generating Station, Units 2 and 3," Docket Nos. 50-361 and 50-362, Southern California Company, January 1982
- "Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station Unit No. 3," Docket No. 50-382, Louisiana Power and Light Company, July 1981
- "Safety Evaluation Report Related to the Operation of Palo Verde Nuclear Generating Station, Units 1, 2 and 3," Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Arizona Public Service Company, October 1984
- "Safety Evaluation Related to Amendment No. 32 to Nuclear Power Facility (NPF)-10 and Amendment No. 21 to NPF-15 for San Onofre Nuclear Generating Station, Units 2 and 3," Docket Nos. 50-361 and

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50-362, Southern California Edison Company, March 1985

- "Safety Evaluation Related to Amendment No. 66 of Facility Operating License No. NPF-6, Arkansas Power & Light Company, Arkansas Nuclear One Unit 2," Docket No. 50-368, May 1985

The COL applicant and ABB-CE will follow the procedures described in "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, Revision 3-P-A, November 1986, and "Reload Data Block Constant Installation Guidelines," CEN-323-P-A, Revision 1-P, December 1986, for all changes to the algorithms, data base constants, and data block constants for the CPCs and CEACs. The staff finds that these procedures have previously been approved by the NRC. The overall CPC/CEAC software implementation, by which the system functional requirements will be translated into modules of machine executable code and the modules integrated into a real-time software system, will be verified through the Phase I and Phase II software verification test. The testing will generate a plant-specific data-base document, appropriate test cases, and acceptable criteria and be documented in test reports. Phase I testing will be performed on the DNBR/LPD calculation systems to verify that CPC/CEAC system software modifications have been properly implemented. Phase II testing on the CPC/CEAC system will verify that CPC and CEAC software modifications have been properly integrated with the CPC and CEAC software and system hardware and to confirm that the static and dynamic operation of the integrated system as modified bears out the predictions of the design analyses.

Testing of the CPC/CEAC software for each license applicant referencing the System 80+ design certification will be considered complete with the formal issuance of (1) a CPC/CEAC database document, (2) the Phase I test report, and (3) the Phase II test report. These documents are plant-specific and will be reviewed individually for each license application referencing the System 80+ design certification. Therefore, DSER Open Item 4.4.4-1 is reclassified as COL Action Item 4.4.4-1.

4.4.5 Thermal-Hydraulic Comparison

In Table 4.3 the thermal-hydraulic design parameters for the System 80+ design are compared with those of the System 80 design. The major differences in the System 80+ design are a 3-percent increase in the rated core power, a decrease of 5.0 °C (9 °F) in the nominal reactor inlet temperature, a 1.5-percent increase in the average mass velocity, and a 1-percent increase in the minimum DNBR in full-power steady-state conditions. The reactor

core heat output increases by 3 percent and the active heat transfer area increases by 3.7 percent. The differences are within 0.5 percent for the coolant volumetric flow rate and average heat flux. Both the System 80 and the System 80+ use the CE-1 DNBR correlation. As described in CESSAR-DC Section 4.4, ABB-CE used thermal design procedures identical to those of the System 80 design, the thermal-hydraulic aspects of which have been approved by the NRC. Since the design parameters for the System 80+ design are within the applicable ranges of the NRC-approved thermal design procedures for System 80, the staff concludes that ABB-CE's approach is acceptable.

4.4.6 Inadequate Core Cooling System Design

ABB-CE describes the inadequate core cooling (ICC) system design in CESSAR-DC Section 7.5.1.1.7 and a response to Three Mile Island (TMI) Action Item II.F.2 in CESSAR-DC Chapter 20. ABB-CE used an inadequate core cooling instrumentation (ICCI) package that the NRC staff had previously approved. The ICCI package consists of resistance temperature detectors (RTDs), pressurizer pressure sensors, core exit thermocouples (CETs), and reactor vessel level monitoring system probes employing the heated junction thermocouple (HJTC) concept. Details of the system are discussed below.

4.4.6.1 Saturation Margin Monitor

The saturation margin monitor (SMM) informs the reactor operator of approaching and existing RCS saturation and existence of core uncover. The SMM inputs include RCS cold-leg and hot-leg temperatures measured by RTDs, the maximum temperature of the top three unheated HJTCs, representative CET temperature, and pressurizer pressure sensors. The unheated HJTC input comes from the output of the HJTC processing units.

4.4.6.2 Core Exit Thermocouple

The CETs measure core heatup by measuring core exit steam temperature. The design of the in-core instrumentation (ICI) system includes Type K (Chromel-Alumel) thermocouples each of the ICI detector assemblies. These CETs monitor the temperature of the reactor coolant as it exits the fuel assemblies. The CETs have a usable temperature range from 0 °C to 1260 °C (32 °F to 2300 °F).

4.4.6.3 Heated Junction Thermocouple System

The HJTC assembly measures reactor coolant inventory above the fuel alignment plate with discrete HJTC sensors. Enclosed in a separator tube, the sensors are at different

Table 4.3 Reactor design comparison

Characteristics	System 80+	System 80
Performance Characteristics		
Reactor core heat output (Mwt)	3,914	3,800
System pressure, kPa (psia)	15.5×10^3 (2,250)	15.5×10^3 (2,250)
Minimum DNBR at steady-state full power	2.00	1.98
Minimum DNBR limit	1.24 (MSCU)	1.24 (SCU)
Critical heat flux correlation	CE-1	CE-1
Coolant Flow		
Total flow rate L per min (gpm)	1.68×10^6 (444,650)	1.69×10^6 (445,600)
Effective flow rate for heat transfer (gpm)	1.63×10^6 (431,300)	1.64×10^6 (432,200)
Average velocity along fuel rods m/s (ft/s)	5.1 (16.7)	5.1 (16.8)
Average mass velocity 10^6 kg/hr-m ² (10^6 lb/hr-ft ²)	13.0 (2.65)	12.8 (2.62)
Coolant Temperature		
Nominal Reactor Inlet °C (°F)	291 (556)	296 (565)
Average Rise in Core °C (°F)	33.9 (61)	32.2 (58)
Heat Transfer 100-percent power		
Active heat transfer surface area m ² (ft ²)	6.6×10^3 (70,960)	6.4×10^3 (68,320)
Average heat flux w/m ² (BTU/hr-ft ²)	5.77×10^5 (183,300)	5.82×10^5 (184,800)
Maximum allowable heat flux w/m ² BTU/hr-ft ²)	1.35×10^6 (429,100)	1.36×10^6 (432,700)
Average linear heat rate kw/m (kw/ft)	17.6 (5.36)	17.7 (5.41)
Peak allowable linear heat generation rate kw/m (kw/ft)	41.3 (12.6)	41.7 (12.7)

levels ranging from the top of the fuel alignment plate to the reactor vessel head and detect the temperature difference between adjacent heated and unheated thermocouples. The HJTC sensor consists of a Chromel-Alumel thermocouple near a heater (or heated junction) and another Chromel-Alumel thermocouple away from the heater (or unheated junction). Two probe assemblies allow

two channels of HJTC instruments. Each HJTC probe assembly has eight HJTCs, a separator tube, a seal plug, and electrical connectors. The eight HJTC sensors are electrically independent. The separator tube creates a collapsed liquid level that the HJTC sensors measure. This collapsed liquid level is directly related to the average

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liquid fraction of the fluid in the reactor head volume above the fuel alignment plate.

ABB-CE states in CESSAR-DC Section 7.5.1 that two design features ensure proper HJTC operation under all thermal-hydraulic conditions. First, each HJTC is shielded to avoid overcooling due to direct water contact during two-phase fluid conditions. The HJTC with the splash shield is referred to as the HJTC sensor. Second, a string of HJTC sensors is enclosed in a tube that separates the liquid and gas phases that surround it.

The ICC sensors send data to

- (1) the discrete indication and alarm system (DIAS) channel P processing equipment for continuous display on the safety-monitoring panel in the main control room,
- (2) the DIAS channel N processing equipment for normal and postaccident monitoring, and
- (3) the data processing system for integration into the cathode ray tube plant process display.

The staff has reviewed the design of the ICC system described in CESSAR-DC Section 7.5.1.1.7 and finds that the ICC system design is consistent with the guidelines in RG 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assist Plant and Environs Conditions During and Following an Accident," and meets the requirements of NUREG-0737, TMI Action Plan Item II.F.2. However, the staff required ABB-CE to complete testing and calibrating the ICC system as part of inspections, tests, analyses, and acceptance criteria (ITAACs). In the DSER, this requirement was designated as DSER Confirmatory Item 4.4.6.3-1. In the response, ABB-CE put the limiting conditions for operation and surveillance requirements in TS 3.3.11 for the ICC system and included the ICC system in ITAACs for the RCS. The staff finds that the ITAACs and TS provide reasonable assurance that the ICC system will be installed and will operate in conformance with the design described in CESSAR-DC Section 7.5.1.1.7. Therefore, the staff concludes that the design of the ICC system is acceptable. Therefore, DSER Confirmatory Item 4.4.6.3-1 is resolved.

4.4.7 Conclusions

The staff reviewed the thermal-hydraulic design of the core for the System 80+ design. The review included the design-basis and steady-state analysis of the core thermal-hydraulic performance and concentrated on the differences between the proposed design and designs that the staff previously reviewed and found acceptable. The staff

concludes that ABB-CE has satisfactorily justified all such differences and has performed thermal-hydraulic analyses using analytical methods, DNBR correlations, and the safety limit DNBR that the staff previously approved. Therefore, the staff concludes that the thermal-hydraulic design of the System 80+ design core has appropriate thermal margin to ensure that SAFDLs are not exceeded during any conditions of normal operation and AOOs, and, thus, conforms to the requirements of GDC 10 and is acceptable. The calculated DNBRs during AOOs are evaluated in Sections 15.1 through 15.3 of this report.

Each COL applicant referencing the System 80+ design certification is responsible for the startup test program to measure and confirm thermal-hydraulic design aspects, in accordance with RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants." However, the CESSAR-DC defines ABB-CE's participation and gives guidelines for the preoperational and initial startup test program. The startup testing program is evaluated in Chapter 14 of this report.

4.5 Reactor Materials

4.5.1 Control Element Drive Mechanism (CEDM) Structural Materials

In its review, the staff evaluated CESSAR-DC Section 4.5.1, following the guidelines of SRP Section 4.5.1. Areas reviewed were material specifications, austenitic stainless steel components and their welding and fabrication into plant systems, inspection of these components, heat treatment of other materials, and cleaning and cleanliness control. DSER open items were resolved as follows:

In CESSAR-DC Section 4.5.1.1, ABB-CE states that Inconel 690 is to be used in lieu of Inconel 600 materials for the motor housing assembly. The staff views the Inconel 690 alloy as the preferred nickel-base alloy in the primary and secondary coolant loops because it is more corrosion-resistant than Inconel 600. Thus, the use of Inconel 690 and its equivalent weld metals (Types 52 and 152) will give reasonable assurance of the material integrity of the components and tubing in contact with reactor coolant and most secondary water chemistries because it is more corrosion resistant. On this basis, DSER Open Item 4.5.1-1 is resolved.

ABB-CE proposes to use American Society for Testing and Materials (ASTM) A-708 for the purpose of maintaining the qualifications of older weld procedures. ASTM A-262, Practice A or E (recommended in RG 1.44, "Control of the Use of Sensitized Stainless Steel") has been used by ABB-CE since the mid-1970s for verifying non-sensitiza-

tion of austenitic stainless steel materials and weldments. ABB-CE qualified weld procedures developed prior to the mid-1970s and used A-708 in their qualification. These weld procedures are still in use. There have been no IGSCC failures involving austenitic stainless steel weldments in ABB-CE's nuclear steam supply system (NSSS) units. In addition, the staff has allowed in SRP Section 4.5.1, Paragraph III.2, ASTM A-708 as an acceptable alternative test for ASTM A-262, Practice A or E. On this basis, DSER Open Item 4.5.1-2 is resolved.

ABB-CE proposes to use Stellite, a cobalt-based alloy, for pins and latches in the CEDM. Radioactivation of cobalt is a concern in current nuclear plants. Cobalt application should be avoided in the System 80+ design to keep radiation doses as low as reasonably achievable. In CESSAR-DC Section 5.2.3.2.2, "Materials of Construction Compatibility with Reactor Coolant," ABB-CE states that cobalt-based alloys will be avoided except where no proven alternative exists. Cobalt-free alloys with the wear and corrosion properties of the Stellite (cobalt-base) type alloys are being developed but have not been fully demonstrated to be as durable as the Stellites. Accordingly, the staff acknowledges that the availability of alternative materials for Stellites is limited at this time. Therefore, DSER Open Item 4.5.1-3 is resolved.

ABB-CE proposes to use Type 304 and 316 austenitic stainless steel. However, these materials are susceptible to intergranular stress corrosion cracking (IGSCC) when the oxygen content of the reactor coolant exceeds 0.010 ppm at temperatures above 200 °F during normal operations. During startup and operation of the System 80+ plant, these temperature and chemical conditions are maintained through specified chemistry control. As allowed in RG 1.44, ABB-CE has taken alternative mitigating approaches, giving reasonable assurance of the integrity of austenitic stainless steel components in contact with reactor coolant. On this basis, DSER Open Item 4.5.1-4 is resolved.

The ferrite content limits for austenitic stainless weld metal in the CESSAR-DC were broader than those in industry guidelines (EPRI, "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991) and staff guidance (USNRC, "Technical Report on Materials Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," NUREG-0313, Revision 2, January 1988). In CESSAR-DC Amendment L, ABB-CE revised the CESSAR-DC to be consistent with industry guidelines and staff guidance. The revision should assure an acceptable level of structural integrity for stainless steel welds over the life of the plant. On this basis, DSER Open Item 4.5.1-5 is resolved.

In CESSAR-DC Section 4.5.1.1, ABB-CE states that martensitic stainless steel (Type 403) will be used for the CEDM housing. ABB-CE has specified the following material and heat treatment for this component in the reactor coolant pressure boundary:

- Material SA-182, Type 403, as modified by ASME Code Case 4-N-11.
- Heat treatment Heat to 1800 °F (982 °C) +/- 25 °F (14 °C), air cool and temper at 1125 °F (607 °C) minimum for 4 hours (Code Case 4-N-11).

This is the only component in the reactor coolant pressure boundary using martensitic stainless steel. For this component, ABB-CE has chosen ASME Code Case N-4-11 material (a modified Type 403 martensitic stainless steel) with the additional requirements of ASME SA-182. This heat treatment should provide a Rockwell "C" hardness of the material in the Rc 25-31 range (according to the ASM Metals Handbook, 8th Edition). ASME Code Case N-4-11 is accepted by RG 1.85, "Materials Code Case Acceptability — ASME Section III, Division 1." This material and specific heat treatment are in accordance with the accepted ASME Code Case, and the material should perform adequately during its service life. On this basis, DSER Open Item 4.5.1-6 is resolved.

In CESSAR-DC Section 4.5.1.1, ABB-CE states that Inconel X-750 (aerial monitoring system (AMS) 5698B and AMS 5699B) will be used. ABB-CE uses these materials for springs that will be subjected to elevated temperatures and must resist relaxation. ABB-CE affirms that prototype testing of the CEDM and extensive previous experience in existing reactors have demonstrated the acceptability of these materials for their intended use. The springs are not part of the primary pressure boundary; therefore, Section III of the ASME Code and RG 1.85 do not require them to be listed as accepted.

AMS 5698 and 5699 (Inconel X-750 materials) are drawn from hot-finished wire rod that has been previously ground or has had surface preparation (other than by pickling) to remove seams or other injurious surface imperfections. The wire is heat-treated at 2100 °F before being reduced to size. The staff finds that these materials should perform their function when supported by the maintenance and inspections prescribed by ABB-CE. On this basis, DSER Open Item 4.5.1-7 is resolved.

In CESSAR-DC Section 4.5.1.3.3, ABB-CE limited the carbon content of austenitic stainless steels to

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0.065 percent as compared to the usual specification limit of 0.08 percent, and imposed controls on welding, fabrication, and water chemistry during startup and operation. These mitigating actions to mitigate the occurrence of IGSCC in these stainless steels have proven successful in operating reactors for more than 15 years and give reasonable assurance of the integrity of components made of austenitic stainless steel. On this basis, DSER Open Item 4.5.1-8 is resolved.

ABB-CE has evaluated CEDM materials by extensively testing a CEDM assembly. The CEDM tested surpassed the lifetime requirements, as described in CESSAR-DC Section 3.9.4.4.1. The design duty or lifetime requirement, as defined in CESSAR-DC Section 3.9.4.1, is a total cumulative CEA travel of 100,000 feet of operation without loss of function (not the 60-year plant life). As stated in CESSAR-DC Section 3.9.4.1, the CEDM is designed to operate without maintenance for a minimum of 1-1/2 years and without replacement of components for a minimum of 3 years. Therefore, the test results of the extensively tested CEDM do not need to be verified to the 60-year plant life.

Over the 60-year plant life, the operational requirement for the System 80+ CEAs, except possibly the lead regulating CEA group, is expected to be less than the 100,000 feet of travel (the tested life). If plants institute daily load cycle operation, the lead regulating CEA group may travel more than 100,000 feet.

Since the regulating CEAs are much lighter than the CEA weight used during accelerated CEDM motor life tests, the System 80+ CEDM motors are expected to be capable of operating regulating CEAs for more than 100,000 feet of cumulative travel. Depending on how much the lead regulating CEA group is used, the CEDM motor for this bank of CEAs may need to be replaced once during the 60-year plant life. Replacing the CEDM motor in case of extreme use is an acceptable approach.

As indicated in CESSAR-DC Section 3.9.4.1, all CEDM pressure boundary components have a design life of 60 years. On this basis, DSER Open Item 4.5.1-9 is resolved.

The staff concludes that the CEDM structural materials are acceptable and meet the applicable requirements of GDC 1, 14, and 26 and 10 CFR 50.55a. This conclusion is based on the following considerations.

ABB-CE has demonstrated that the properties of materials selected for the CEDM components exposed to reactor coolant satisfy Appendix I to Section III, and Parts A, B, and C of Section II, of the ASME Boiler and Pressure

Vessel Code. ABB-CE selected all materials in this system for their compatibility with the reactor coolant. Materials were extensively tested in a CEDM assembly that exceeded lifetime requirements. Moreover, all materials have served satisfactorily in many other NRC-licensed operating reactors for more than 20 years.

The mechanical properties of structural materials selected by ABB-CE for control rod system components exposed to the reactor coolant conform with the staff position that the yield strength of cold worked austenitic stainless steel should not exceed 90 Ksi (620.5 MPa). ABB-CE has met the guidelines of RG 1.85, "Code Case Applicability — ASME Section III Materials," by using construction materials that are approved by ASME Code cases.

In addition, the controls imposed on the austenitic stainless steel of the mechanisms conform to the recommendations of RG 1.33, "Control of Ferrite Content in Stainless Steel Weld Metal," and RG 1.44, "Control of the Use of Sensitized Stainless Steel." The controls imposed on the austenitic stainless steels of the system will satisfy the requirements of the materials specifications. All raw austenitic stainless steel, both wrought and cast, used in fabricating the CEDM structural components will be supplied in the solution-annealed condition. ABB-CE avoids sensitization by forbidding heat treatment in the temperature range of 800-1500 °F (427-816 °C). Martensitic stainless steels will be tempered in accordance with the staff's positions, except for an ASTM A-276, Type 440C ball in the vent valve on top of the CEDM. This component is compressively loaded. The yield strength of this material exceeds 90 Ksi (620.5 MPa), but corrosion has not occurred in operating reactors with extensive operating experience. All materials selected for the control rod drive mechanism components conform with the applicable Code case listed in RG 1.85.

Fabrication and heat-treatment practices specified in accordance with these recommendations give added assurance that stress corrosion cracking will not occur during the design life of the components. The compatibility of all materials in the control rod system in contact with the reactor coolant satisfies the criterion of Subarticles NB-2120 and NB-3120 of the Code.

Cleaning and cleanliness control are in accordance with American National Standards Institute (ANSI)/ASME NQA-2-1983, "Quality Assurance Requirements for Nuclear Power Plants," and RG 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The staff has previously reviewed ANSI/ASME NQA-2-1983

and finds it acceptable. The cleaning and cleanliness control specified will adequately control contamination of components during fabrication, shipment, and storage.

Conformance with (1) the codes, standards, and RGs and (2) the staff's positions on the allowable maximum yield strength of cold-worked austenitic stainless steel and on the tempering temperatures for martensitic stainless steels satisfies the requirements of GDC 1, 14, and 26 and 10 CFR 50.55a.

4.5.2 Reactor Internals Materials

The staff reviewed CESSAR-DC Section 4.5.2 in accordance with SRP Section 4.5.2. Areas reviewed were material specifications, control of welding of materials used for reactor internals, nondestructive examination of wrought seamless tubular products and fittings, austenitic stainless steel components, the welding, fabrication, and inspection of reactor internals, and heat treatment of other materials. DSER open items were resolved as follows:

ABB-CE proposes to use Stellite, a cobalt-based alloy, as a hard-facing material. As discussed in Section 4.5.1 of this report, ABB-CE states that cobalt-based alloys will be avoided unless no proven alternative exists. Cobalt-free alloys with the wear and corrosion properties of the Stellite (cobalt-base) alloys are under development, but have not been fully demonstrated to be as usable as the Satellites. Accordingly, the staff acknowledges that at this time the availability of alternative materials for Satellites is limited. On this basis, DSER Open Item 4.5.2-1 is resolved.

As discussed in CESSAR-DC Section 4.5.1, ABB-CE proposes to allow the continued reference to ASTM A-708 for the purpose of maintaining the qualifications of older weld procedures. ASTM A-262, Practice A or E (recommended in RG 1.44), has been used by ABB-CE since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. ABB-CE qualified weld procedures developed prior to the mid-1970s and used A-708 in their qualification. These weld procedures are still in use. There have been no IGSCC failures involving austenitic stainless steel weldments in ABB-CE's NSSS units. In addition, the staff allowed in SRP Section 4.5.1, Paragraph III.2, ASTM A-708 as an acceptable test for ASTM A-262, Practice A or E. On this basis, DSER Open Item 4.5.2-2 is resolved.

ABB-CE proposes to use Type 304 austenitic stainless steel. However, this material is susceptible to IGSCC when the oxygen content of the reactor coolant exceeds 0.010 ppm at temperatures above 200 °F during normal operations. During startup and operation of the ABB-CE

NSSS, these temperature and chemical conditions are maintained through specified chemistry control. On this basis, DSER Open Item 4.5.2-3 is resolved.

In CESSAR-DC Section 4.5.2.1, ABB-CE indicates that Inconel 690 will be used in lieu of 600 to fabricate the flow skirt. The staff views the Inconel 690 alloy as the preferred nickel-base alloy in the primary and secondary coolant loops because it is more corrosion-resistant than Inconel 600. The use of Inconel 690 and the equivalent weld metals (Types 52 and 152) will give reasonable assurance of the material integrity of the components in contact with reactor coolant. On this basis, DSER Open Item 4.5.2-4 is resolved.

As discussed in Section 4.5.1 of this report, the ferrite content limits for austenitic stainless steel castings and weld metal given in the CESSAR-DC were broader than those in industry guidelines and staff guidance. In Amendment N to the CESSAR-DC, ABB-CE revised the ferrite content of weld metal to be consistent with industry guidelines. In Amendment Q to the CESSAR-DC, ABB-CE reduced the upper limit of ferrite content in stainless steel castings with normal operating temperatures above 260 °C (500 °F) (as expressed in ferrite number (FN)) to 20 to mitigate the effects of thermal embrittlement over the life of the castings in service. This reduction will offer reasonable assurance of an acceptable level of fracture toughness for stainless steel castings over the life of the plant. On this basis, DSER Open Item 4.5.2-5 is resolved.

In CESSAR-DC Section 4.5.2.1, ABB-CE states that precipitation-hardened stainless steel will be used. The precipitation-hardened stainless steel used in the reactor internals is SA 453 Grade 660 or SA 638 Grade 660. ABB-CE has specified the following heat treatment for these materials:

- Solution Treatment: 1650 ± 25 °F (899 °C ± 14 °C), for 2 hours minimum, oil or water quench
- Hardening Treatment: 1350 ± 25 °F (732 °C ± 14 °C), for 16 hours, air cool

Part A of the 1992 ASME Code Section II, "Materials," indicates the hardening temperature for both SA-453 and SA-638 should be within the range of 1300-1400 °F (704-760 °C). These heat treatments will offer these materials acceptable mechanical and corrosion-resistant properties. On this basis, DSER Open Item 4.5.2-6 is resolved.

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In CESSAR-DC Section 4.5.2.3.1.4, ABB-CE limits the carbon content of austenitic stainless steels to 0.065 percent compared to the usual specification limit of 0.08 percent and imposes controls on welding, fabrication, and water chemistry during startup and operation. These actions to mitigate the occurrence of IGSCC in these stainless steels have proven successful in operating reactors for more than 15 years and give reasonable assurance of the integrity of these components. On this basis, DSER Open Item 4.5.2-7 is resolved.

The staff concludes that the materials used for the construction of the reactor internals and core support structures are acceptable and meet the requirements of GDC 1 and 10 CFR 50.55a. This conclusion is based on the following considerations.

ABB-CE satisfies the requirements of GDC 1 and 10 CFR 50.55a with respect to assuring that the design, fabrication, and testing of the materials used in the reactor internals and core structures support meet standards of high quality and are adequate for structural integrity. The controls imposed components made of austenitic stainless steel satisfy the recommendations of RG 1.31.

The materials used for constructing the components of the reactor internals and core support structures have been identified by specification and conform with the requirements of NG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code. In addition, ABB-CE conforms with the guidelines of RG 1.85 by using construction materials approved for use by ASME Code cases. As proven by satisfactory performance, the specified materials are compatible with the expected environment and corrosion is expected to be negligible.

The controls imposed the reactor coolant chemistry offer reasonable assurance that the reactor internals and the core support structures will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of component structural integrity.

The material selection, fabrication practices, examination and testing procedures, and control practices with conforming the ASME Code and RGs 1.31 and 1.44, offer reasonable assurance that the materials used for the reactor internals and core support structures will not deteriorate in service. Conformance with requirements of the ASME Code and the recommendations of these RGs satisfies the chemical and material requirements of GDC 1 and 10 CFR 50.55a.

4.6 Functional Design of Reactivity Control Systems

The functional design of the reactivity control systems for the System 80+ standard plant is within the scope of design certification, and the staff reviewed CESSAR-DC to confirm that the design has the capability to satisfy the various reactivity control conditions for all modes of plant operations.

These control conditions are the following:

- (1) the capability to operate in the unrodded, critical, full-power mode throughout plant life,
- (2) the capability to vary power level from full power to hot shutdown and have power distributions within acceptable limits at any power level, and
- (3) the capability to shut down the reactor to mitigate the effects of postulated events discussed in Chapter 15 of this report.

The reactivity control systems for the facility are the control rod drive system (CRDS), the safety injection system (SIS), and the CVCS.

The CRDS contains magnetic-jack CEDMs. When electrical power is removed from the coils of the CEDM, the armature springs automatically disengages the holding latches from the CEDM drive shaft, allowing insertion of the CEAs and the PSCEAs by gravity. There are 68 full-strength CEAs and 25 PSCEAs. The regulating CEA groups may be used to compensate for changes in reactivity associated with power-level changes and power distribution, variations in moderator temperature, or changes in boron concentration. Refer to Sections 3.9.4 and 4.3 of this FSER for further discussion of this feature. The PSCEA consists of an Inconel 625 tube loaded with Inconel 625 bars over the full active length. The PSCEAs, which have lower worth than the full-strength CEAs, control reactivity and axial power shape during power operations.

The SIS automatically injects boric acid into the RCS on receipt of a safety injection actuation signal. The SIS pumps take suction from the in-containment refueling water storage tank. The SIS is discussed further in Section 6.3 of this report.

The CVCS is a non-safety-grade system designed to control slow or long-term reactivity changes, such as those

caused by fuel burnup and variations in coolant temperature, and the xenon concentration. The CVCS controls reactivity by adjusting the dissolved boron concentration in the RCS. The boron concentration is controlled to obtain optimum CEA positioning, to compensate for reactivity changes during startup, load following (changes in reactor power level), and shutdown, and to provide shutdown margin for maintenance and refueling operations or emergencies. The boric acid concentration in the RCS is controlled by the charging and letdown portions of the CVCS.

The CVCS can be used to maintain reactivity within the required bounds by means of the automatic makeup system which replaces minor coolant leakage without significantly changing the boron concentration in the RCS system. Dilution of the RCS boron concentration is required to compensate for reactivity losses from fuel depletion. Dilution is accomplished by manual operation of the CVCS. The CVCS is discussed further in Section 9.3.4 of this report.

The concentration of boron in the RCS is changed manually under the following operating conditions:

- (1) Startup - boron concentration is decreased to compensate for moderator temperature and power increase.
- (2) Load follow - boron concentration is increased or decreased to compensate for moderator temperature and power changes during load changes, and for xenon transients following load changes.
- (3) Fuel burnup - boron concentration is decreased to compensate for burnup.
- (4) Cold shutdown - boron concentration is increased to compensate for increased moderator density due to cooldown.

Soluble absorber concentration changes are used to control slow reactivity changes, but if necessary, CEAs can also be inserted to accommodate such changes, although insertion is used mainly to mitigate AOOs (the analysis assumes a single malfunction, such as a stuck rod). In either case, fuel design limits will not be exceeded. The soluble boron control is capable of keeping the core subcritical during cold shutdown, in conformance with GDC 26, "Reactivity Control System Redundancy and Capability."

The CEAs are the primary shutdown mechanism for normal operation, accidents, and transients. They are inserted automatically in accident and transient conditions.

Concentrated boric acid solution is injected by the SIS in the event of a LOCA, steamline break, loss of normal feed-water flow, steam generator tube rupture, or CEA ejection, thereby complying with GDC 20, "Protection System Functions," which requires automatic protective systems (1) to ensure that SAFDLs are not exceeded and (2) to sense accident conditions and actuate safety related systems and components.

The ability of the CEAs to have their position changed is tested quarterly during power operation. At every refueling shutdown, the CEAs are stepped over their entire range of movement and are drop-tested to demonstrate their ability to drop in the required time. The CEAs are designed such that a single failure will not result in loss of the protection system, and removing a channel or component from service will not result in a loss of redundancy. These matters are discussed further in Section 7.2 of this report. The provisions for periodic testing, reliability, and redundancy conform to the requirements of GDC 21, "Protection System Reliability and Testability."

Failure of electrical power to any CEDM will result in the insertion of the CEA. Analysis of accidental withdrawal of a CEA was found to have acceptable results, as discussed in Section 15.2.4 of this report. The requirements of GDC 23, "Protection System Failure Modes," and GDC 25, "Protection System Requirements for Reactivity Control Malfunctions," are thus met.

The reactivity control system functional design meets the requirements of GDC 21, 23, 25, 26, and 27 with respect to its reliability and testability, fail-safe design, malfunction protection design, redundancy and capability, and combined systems capability, and is, therefore, acceptable.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The ABB-CE reactor is a pressurized-water reactor (PWR) with two coolant loops. The reactor coolant system (RCS) circulates water in a closed cycle, removing heat from the reactor core and internal components and transferring it to a secondary system. The steam generators (SGs) serve as the interface between the reactor coolant (primary) system and the main steam (secondary) system. The SGs are vertical U-tube heat exchangers with an integral economizer in which heat is transferred from the reactor coolant to the main steam system. Reactor coolant is prevented from mixing with the secondary steam by the SG tubes and the SG tubesheet, making the RCS a closed system, thus forming a barrier to the release of radioactive materials from the core of the reactor to the secondary system and containment building.

The major components of the system are the reactor vessel; two parallel heat transfer loops, each containing one SG and two reactor coolant pumps; a pressurizer connected to one of the reactor vessel hot legs; and associated piping. All components are located inside the containment building.

System pressure is controlled by the pressurizer, where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the pressurizer, or is condensed by the pressurizer spray to limit pressure variations caused by contraction or expansion of the reactor coolant. The average temperature of the reactor coolant varies with power level, and the fluid expands or contracts, changing the water level in the pressurizer.

The charging pumps and letdown control valves in the chemical and volume control system (CVCS) are used to maintain a programmed level of water in the pressurizer. A continuous but variable letdown purification flow is maintained to keep the RCS chemistry within prescribed limits. A charging nozzle and a letdown nozzle are provided on the reactor coolant piping for this operation. The charging flow is also used to alter the boron concentration or correct the chemical content of the reactor coolant.

Other RCS penetrations are the pressurizer surge line in one hot leg; the four direct vessel injection (DVI) nozzles for the safety injection system (SIS); the two return nozzles to the shutdown cooling system, one in each hot leg; two pressurizer spray nozzles; vent and drain connections; and sample and instrument connections.

Overpressure protection for the reactor coolant pressure boundary (RCPB) is provided by four spring-loaded

American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) safety valves connected to the top of the pressurizer. These valves discharge to the in-containment refueling water storage tank (IRWST), where the steam is released under water to be condensed and cooled. If the steam discharge exceeds the capacity of the IRWST, it is vented to the containment atmosphere.

Overpressure protection for the secondary side of the SGs is provided by spring-loaded ASME Code safety valves located in the main steam system upstream of the steamline isolation valves.

Components and piping in the RCS are insulated with a material compatible with the temperatures involved, in order to reduce heat losses and protect personnel from high temperatures.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With Codes and Code Cases

5.2.1.1 Compliance With 10 CFR 50.55a

The components of the RCPB are defined by the rules of 10 CFR 50.55a, "Codes and Standards," and have been properly classified in CESSAR-DC Table 5.2-1 as ASME Code Section III, Class 1 components. These components are designated as Quality Group A in CESSAR-DC Table 3.2-1 and are in conformance with Regulatory Guide (RG) 1.26, "Quality Standards and Records." In CESSAR-DC Table 1.8-6, ABB-CE identifies the 1989 edition of the ASME Code and addenda for System 80+ design. The specific edition and code has been endorsed by 10 CFR 50.55a and is, therefore, acceptable. The specification of the ASME Code is considered to be Tier 1 information. However, specification of the code editions and addenda is considered to be Tier 2 information and, as such, is subject to change by a COL applicant through a 50.59-like process. Therefore, it will be required that each applicant for a combined license (COL) that references the CESSAR-DC identify the ASME Code edition to be used in the construction of the RCPB ASME Class 1 components if it differs from the edition specified in the CESSAR-DC. In Amendment U to CESSAR-DC Section 5.2.1.1, ABB-CE committed that the COL applicant will identify the ASME Code edition and addenda other than those specified in CESSAR-DC Table 1.8-6, and that the Code and addenda shall have been endorsed in 10 CFR 50.55a. This is COL Action Item 5.2.1.1-1. In addition, those RCPB components that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified Quality Group B in accordance with the guidance in Position C.1

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of RG 1.26 and are constructed as ASME Section III, Class 2 components.

Since this position is not totally consistent with ASME Code, Section III, Subsection NCA-1140, "Use of Code Editions, Addenda, and Cases," the following comparison is made:

NCA-1140 Rules

NCA-1140(a)(1) - Under the rules of this Section, the owner or its designee shall establish the Code edition and addenda to be included in the Design Specifications. All items of a nuclear power plant may be constructed to a single Code edition and addenda, or each item may be constructed to individually specified Code editions and addenda.

NCA-1140(a)(2) - In no case shall the Code edition and addenda dates established in the Design Specifications be earlier than 3 years prior to the date that the nuclear power plant construction permit application is docketed.

NCA-1140(b) - Code editions and addenda later than those established by (a) above may be used by mutual consent of the owner or its designee and certificate holder. For Division 2 design and construction, the consent of the Designer shall also be obtained. Specific provisions within an edition or addenda later than those established in the design specifications may be used, provided that all related requirements are met.

NCA-1140(c) - Code cases are permissible and may be used beginning with the date of approval by the ASME Council (and the American Concrete Institute for Division 2 design and construction). Only Code cases that are specifically identified as being applicable to this Section may be used.

Design Certification Position

(1) The specific Code edition and addenda shall be specified by ABB-CE in the CESSAR-DC during design certification. The COL applicant may update the Code edition or addenda (or portions thereof) referenced in the CESSAR-DC using a 10 CFR 50.59-like process without prior NRC approval unless the proposed change involves a change to the certified design or an unreviewed safety question. The COL applicant may update the referenced edition or addenda for all items of a nuclear power plant or a specific item if the construction practices (including fabrication, inspection, and examination methods) are not compatible with the design requirements. All changes to the referenced edition and addenda shall be documented and maintained by the COL applicant and available for audit.

(2) The specific Code edition and addenda are required to be established during design certification except for Section XI requirements related to inservice inspection (ISI), inservice testing, and system pressure tests, which shall meet the requirements of 10 CFR 50.55a(g). Further discussions and any exceptions to 10 CFR 50.55a(g) are noted in Sections 5.2.4 and 6.6 of this report.

(3) This is acceptable subject to the conditions noted in position (1) above.

(4) As discussed in Section 5.2.1.2 of this report, only those Code cases identified in RG 1.84, "Code Case Acceptability in ASME Section III - Design and fabrication," Rev. 28 or RG 1.85, "Code Case Acceptability in ASME Section III - Materials," Rev. 28, as specified in the CESSAR-DC may be used. The COL applicant may

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submit for staff review and approval future Code cases that are endorsed in future revisions of RG 1.84 and 1.85 with its COL application provided these cases do not involve a change to the certified design or an unreviewed safety question.

NCA-1140(d) - Code cases may be used by mutual consent of the owner or its designee, and the certificate holder on or after the date permitted by (c) above. For Division 2 design and construction, the consent of the designer shall also be obtained.

(5) Code cases to be used in the design of the standard plant must be identified by ABB-CE in the CESSAR-DC during design certification.

NCA-1140(e) - Existing materials previously produced and certified in accordance with Code editions and addenda earlier than the one specified for construction of an item may be used, provided all of the following requirements are satisfied.

(i) The material (NCA-1220) meets the applicable requirements of a material specification permitted by paragraph 2121 of the applicable subsection of the Section III edition and addenda specified for construction.

(ii) The material meets all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and addenda specified for construction.

(iii) The material was produced under the provisions of a Quality System Program that had been accepted by the Society or qualified by a party other than the Society (NCA-3820), in accordance with the requirements of the latest Section III Edition and Addenda issued at the time the material was produced. Material exempted from portions of the provisions of NCA-3800 by paragraph 2610 of the applicable Subsection of Section III may be used, provided the requirements of (i) and (ii) above are met.

(6) Does not apply to design certification.

NCA-1140(f) - Code editions, addenda [including the use of specific provisions of editions addenda permitted by (b) and (e) above], and Cases used shall be reviewed by the owner or its designee for acceptability to the regulatory and enforcement authorities having jurisdiction at the nuclear power plant site.

(7) The specific Code edition and addenda shall be specified by ABB-CE in the CESSAR-DC during design certification. Use of later editions and addenda (or portions thereof) and Cases not approved during design certification shall be reviewed by the COL applicant or its designee for acceptability to the regulatory and enforcement authorities having jurisdiction at the nuclear power plant site.

Conclusion

The staff concludes that the construction of all ASME Class 1, 2, and 3 components and their supports will

conform to the appropriate ASME Code editions and addenda as allowed and modified by the Commission's regulations and that component quality will be commensurate with the importance of the safety function of all such

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components and their supports. This constitutes an acceptable basis for satisfying General Design Criterion (GDC) 1, "Quality Standards and Records," and is acceptable.

5.2.1.2 Applicable Code Cases

In amendments to the CESSAR-DC, ABB-CE revised Section 5.2.1.2 to state that the RCPB components shall be fabricated in accordance with the ASME Code, Section III and that code case interpretations applied to these components are specified in Table 1.8-7. The specific Code Cases whose requirements will be applied in the design of ASME Section III, Division 1, Class 1, Class 2, Class 3, and Class MC components are in accordance with the rules of 10 CFR 50, Section 50.55a and the guidance provided in RGs 1.84 and 1.85, or have been specifically approved by the staff as part of its design certification review. None of the specified Code Cases involve ISI, accordingly, RG 1.147 is not relevant.

Code Cases other than those specified above will be identified by the COL applicant. Also, unless stated otherwise in the COL application, the COL applicant is committed to comply with RGs 1.84, 1.85 and 1.147 in effect at that time in determining suitable ASME Code cases. A code case not included in these RGs may be used with specific authorization from the Commission under 10 CFR Part 50, Section 50.55a. On this basis, DSER Open Item 5.2.1.2-1 is resolved.

The staff concludes that compliance with the ASME Code cases listed in RGs 1.84, and 1.85, or as approved under 10 CFR Part 50, Section 50.55a will result in a level of quality that is commensurate with the importance of the safety function of the RCPB and constitutes an acceptable basis for satisfying the requirement of GDC 1 and is acceptable.

5.2.2 Overpressure Protection

The staff reviewed the design for overpressure protection for the System 80+ design described in CESSAR-DC Sections 5.2.2, 5.4.10, and 5.4.13, and CESSAR-DC Appendix 5A in accordance with Standard Review Plan (SRP) Section 5.2.2. Overpressure protection for the RCPB is provided by pressurizer safety valves, SG safety valves, and relief valves in the shutdown cooling system (SCS), in combination with the action of the reactor protection system. The combination of these features provides overpressure protection as required by GDC 15, "Reactor Coolant System Design," Section III of the ASME Code, "Rules for Construction of Nuclear Power Plant Components," and 10 CFR Part 50, Appendix G,

"Fracture Toughness Requirements." These measurements ensure RCPB overpressure protection for power operation and low-temperature operation (startup and shutdown). Both modes of overpressure protection are discussed below.

5.2.2.1 Overpressure Protection During Power Operation

For power operation, the overpressure protection design maintains primary and secondary operating pressures within 110 percent of design by means of 4 pressurizer safety valves, 20 SG safety valves, and the reactor protection system. The secondary safety valves are sized to pass a total steam flow of 8.6×10^6 kg/hr (19×10^6 lbm/hr) at the safety valves set pressures of at least 8,275 kPa (1,200 psia) plus 3-percent accumulation pressure to limit SG pressure to less than 110 percent of SG design pressure for design transients. The reactor trips at a pressurizer pressure of 16,780 kPa (2,434 psia), and the pressurizer safety valves are designed to lift at a pressure of 17,200 kPa (2,500 psia), which is the RCS design pressure.

The design-basis event for sizing the pressurizer safety valves is a loss of turbine generator load without a simultaneous reactor trip. In the analysis submitted, the reactor trip is assumed to occur on the second safety-grade signal (generated by the core protection calculator at pressurizer pressure of 16,990 kPa (2,464 psia)). (The first safety-grade signal generated by the reactor protection system is due to a pressurizer pressure signal of 16,780 kPa (2,434 psia)). No control-grade reactor trip signals are credited. Also, no credit is taken for letdown, charging, pressurizer spray, turbine bypass, or feedwater addition after the turbine trip. The initial RCS and main steam system are at the maximum rated output of 3,817 MWt plus a 2-percent uncertainty. The moderator and Doppler coefficients used for the analysis maximize the power and pressure excursion. An upper limit on instrument error in pressurizer pressure measurement is used in the analysis. This delays the reactor trip, which, in turn, provides additional conservatism in sizing the pressurizer safety valves.

The assumptions used for the analysis meet acceptance criterion II.2.A.c(3) of SRP Section 5.2.2, which states that the reactor trip should be initiated by the second safety-grade signal from the reactor protection system. Under the assumptions of this design, the minimum capacity of the four pressurizer safety valves is determined to be 209,090 kg/hr/valve (460,000 lbm/hr/valve). The calculated peak primary and secondary system pressures are less than 110 percent of design pressure, required by GDC 15, during the limiting case of the loss of load transient.

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Further, ABB-CE increased the maximum rated power from 3,817 MWt to 3,931 MWt (a 3-percent increase in the rated power), and assessed the applicability of the analysis to the upgraded power conditions. During a loss-of-load event, the 3-percent increase in power level will cause the primary pressure to increase at a faster rate and result in an earlier reactor trip, which rapidly reduces the insurge into the pressurizer, and thus results in a lower pressure increase. In addition, the upgraded power design increased the minimum capacity of the safety valves by 14 percent to 238,636 kg/hr/valve (525,000 lbm/hr/valve). Because of a combination of the earlier reactor trip and the increased safety valve capacity, ABB-CE asserted that the consequences of the loss-of-load event was less severe at the 3,931 MWt power conditions. Without a quantitative analysis, it was not obvious to the staff that the higher pressure increase during a loss-of-load event at a higher initial power level would be offset by larger primary safety valves. As a result, the staff required ABB-CE to perform an analysis for the loss-of-load event based on a power level of 3,931 MWt, and present the results of the limiting case in CESSAR-DC Section 5.2.2 for the staff to review. This was designated as DSER Confirmatory Item 5.2.2.1-1.

In Amendment U to the CESSAR-DC, ABB-CE provided the analysis in Appendix 5A, which shows that the calculated peak primary and secondary pressures resulting from a loss-of-load event are within 110 percent of the design pressures. On this basis, DSER Confirmatory Item 5.2.2.1-1 is resolved.

In CESSAR-DC Section 15.2.8, ABB-CE presented an analysis of extended safety valve blowdown for the System 80+ pressurizer safety valves. The analysis shows that for the worst-case feedwater line break, the water level in the pressurizer will not reach the inlet nozzle of the pressurizer safety valve. The analyses assume an 18.5-percent blowdown (valve setpoint of 14,300 kPa (2,070 psia)) below the nominal set pressure of 17,500 kPa (2,540 psia) for the pressurizer safety valves instead of the 5 percent specified by the ASME Code. The lower setpoint assumed in the analyses results in increased blowdown and maximizes the water level increase in the pressurizer. ABB-CE determined that water will not be relieved through the safety valves. The design specification and procurement process ensure that the System 80+ pressurizer safety valves are consistent with the valves that were previously tested. The safety valves are designed in accordance with ASME Code, Section III, and periodic testing and inspection are performed in accordance with ASME Code, Section XI, "Rules of Inservice Inspection of Nuclear Power Plant Components."

The overpressure protection design for the System 80+ plant at power operating conditions complies with the guidelines of SRP Section 5.2.2 and the requirements of GDC 15 and, therefore, is acceptable for design certification.

5.2.2.2 Overpressure Protection During Low-Temperature Operation

Guidelines in SRP Section 5.2.2 state that the system for overpressure protection during low-temperature phases of plant operation should be designed in accordance with the requirements of Branch Technical Position (BTP) RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures."

The low-temperature overpressure protection (LTOP) system for the System 80+ design is provided by the spring-loaded liquid-relief valves in the SCS. One SCS liquid relief valve is provided in each of the two SCS pump suction lines. These two valves are set at a pressure low enough to prevent violation of the 10 CFR Part 50, Appendix G heatup and cooldown curves should a pressure transient occur during low-temperature operations. For LTOP considerations, two types of events are considered as the design-basis events. These events are (1) the mass addition transient caused by charging and safety injection (SI) flows following an inadvertent SI actuation and (2) the heat addition transient caused by the restart of a reactor coolant pump. ABB-CE determined the pressure setpoint and flow capacity for the SCS relief valves based on the mass addition transient, which has been demonstrated as the limiting case, and which results in the highest pressure increase. The mass addition transient analysis was performed assuming simultaneous operation of four SI pumps and one charging pump with the letdown system isolated. All pressurizer heaters were assumed to be operating from a water solid condition to maximize the pressure increase. The result shows that the peak pressure is 4,275 kPa (620 psia) for a relief valve with the pressure setpoint of 3,760 kPa (545 psia) and flow capacity of 19,000 L/min (5,000 gpm) of water. ABB-CE specified 19,000 L/min (5,000 gpm) for each of two valves as a rated relief capacity. This rated relief capacity will meet the required relief flow for the worst transient.

LTOP enable temperatures for initiation of the LTOP system can be determined by following BTP RSB 5-2 guidance for LTOP. According to the BTP 5-2 guidance, the enable temperature is the water temperature corresponding to metal temperature of at least reference nil-ductility transition temperature (RT_{NDT}) + 50 °C (90 °F) at vessel beltline location (either 1/4 t or 3/4 t) that is controlling in the Appendix G (to section III of the ASME Code) limit calculations. Conforming with BTP 5-2

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guidance, ABB-CE has determined the disable temperatures for the plant heatup as a function of heatup rates and the enable temperature for the controlling cooldown. However, ABB-CE proposed LTOP disable and enable temperatures defined by intersections of the 10 CFR Part 50, Appendix G heatup and cooldown curves and the pressurizer safety valve setpoint of 17,200 kPa (2,500 psia). The proposed LTOP temperature limits are specified in Technical Specifications (TS) 3.4.3 and 3.4.11. Since the proposed disable and enable temperatures specified in the TS exceed the required temperatures in accordance with BTP RSB 5-2 for LTOP with sufficient margin in overpressure protection, the staff determined that the proposed LTOP temperature limits are acceptable.

For temperatures above the LTOP enable temperature, overpressure protection is provided by the pressurizer safety valves. Before entering the low-temperature region for which LTOP is necessary, administrative controls require operators to decrease the RCS pressure below the maximum pressure allowable for SCS operation. The SCS will be aligned whenever the RCS is at low temperatures and the reactor vessel head is secured, or until an adequate vent has been established.

System design criteria required by the staff include the following: the mitigating system must meet single-active failure criteria; the system must be capable of being tested; and the system must be capable of functioning following loss-of-offsite power. ABB-CE has met all the design criteria for LTOP. This provides assurance that the temperature-pressure limit presented in Appendix G of 10 CFR Part 50 will not be exceeded during any transients.

DSER Open Item 5.2.2.2-1, regarding the use of cobalt containing alloys, such as Stellite, is resolved as follows:

Stellites are cobalt alloys. Activation of cobalt is a concern relating to the radioactivity in current nuclear plants. Therefore, ABB-CE should avoid the use of cobalt in order to maintain the philosophy that exposure to radioactivity should be kept as low as reasonably achievable (ALARA). In CESSAR-DC Section 5.2.3.2.2, "Materials of Construction Compatibility With Reactor Coolant," ABB-CE states that alloys containing cobalt will be avoided, except in cases where no proven alternative exists. On this basis, DSER Open Item 5.2.2.2-1 is resolved. Although cobalt-free alloys with the wear and corrosion properties of the Stellite (cobalt) alloys are under development, they have not been fully demonstrated to have the usability of the Stellites at this time. The NRC staff encourages the COL applicant to monitor the continuing development of cobalt-free hardfacing alloys in nuclear

power plant systems to reduce future radiation exposure of personnel.

The COL applicant should determine the LTOP enable temperature based on plant-specific material properties and pressure-temperature limit curves. This is COL Action Item 5.2.2.2-1.

5.2.2.3 Pressurized Thermal Shock

Pressurized thermal shock (PTS) events are system transients in a pressurized-water reactor that can cause severe overcooling followed by immediate repressurization to a high level. The thermal stresses, caused when the inside surface of the reactor vessel cools rapidly, combine with the pressure stresses to increase the potential for fracture if an initiating flaw is present in low-toughness material. This material may exist in the reactor vessel beltline, adjacent to the core, where neutron radiation gradually embrittles the material over time. The chemical composition of the steel is an important determining factor regarding the degree of embrittlement. In 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" (56 FR 22300; May 15, 1991), the staff establishes a PTS screening criterion (RT_{PTS}) below which no additional action is required for protection from PTS events.

In response to Q252.13, ABB-CE stated that the design conforms to the PTS screening criterion. The ABB-CE assessment is based on the assumed beltline forging material properties (0.06 weight-percent copper, 1.00 weight-percent nickel, and -12 °C (10 °F) initial RT_{NDT}) and 6×10^{19} neutrons/cm² end-of-life (EOF) fluence (60 years).

The reactor vessel beltline materials are acceptable and comply with the requirements of 10 CFR 50.61. DSER Open Item 5.2.2.3-1 is resolved as follows:

In Amendment N, ABB-CE modified the CESSAR-DC to state that the PTS calculation margin used was 27 °C (50 °F) for base metal.

The staff has performed a statistical analysis of all the plate surveillance data that were received from the licensees. The surveillance data are maintained in the Power Reactor — Embrittlement Database (PR-EDB). Surveillance data received by the NRC from the licensees after the issuance of RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Revision 2, indicate that the standard deviation (σ) for the increase in reference temperature for the plate material should be 13 °C (26 °F). This would result in a margin

of 28.9 °C (52 °F). The staff has raised this issue in the PTS review of Indian Point 3 and Beaver Valley 1. Since the staff's statistical analysis indicates that the standard deviation for the increase in reference temperature is greater than 9.5 °C (17 °F), the licensee's use of 9.5 °C (17 °F) as the amount of margin may not ensure that the RT_{PTS} value is less than the PTS screening criteria at the end of a license. A staff contractor is currently reviewing the data base to update the methodology for calculating RT_{PTS} . When this review is completed, a margin term different than 9.5 °C (17 °F) may be necessary to ensure that the RT_{PTS} value is less than the PTS screening criterion at the end of a license.

The staff believes that ABB-CE's approach of using a margin of 27.8 °C (50 °F) rather than 28.9 °C (52 °F) has enough conservatism to provide adequate assurance of fracture toughness in a PTS event for its design life. Information is being developed to improve the accuracy of calculating RT_{PTS} . This issue is further addressed in the resolution of DSER Open Item 5.3.2-3. On this basis, DSER Open Item 5.2.2.3-1 is resolved.

The COL applicant should verify that plant-specific material properties and EOF fluence (60 years) are within the limits assumed in the CESSAR-DC analysis. This is COL Action Item 5.2.2.3-1.

5.2.3 Reactor Coolant Pressure Boundary Materials

The staff reviewed the RCPB materials described in CESSAR-DC Section 5.2.3 in accordance with SRP Section 5.2.3. The areas reviewed were materials specifications, compatibility of materials with reactor coolant, fabrication and processing of ferritic materials, and the fabrication and processing of stainless steels. The staff's open items are resolved as follows:

ABB-CE took exception to the recommendations in Position C.2 of RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," for controls imposed on preheat temperatures for welding ferritic steels. These controls would provide reasonable assurance that components made from low-alloy steels will not crack during fabrication and minimize the possibility of subsequent cracking from hydrogen in the weldment. ABB-CE, in a letter of November 24, 1992, stated that its basis for taking exception to Position C.2 was Westinghouse Topical Report WCAP-8678, "Effect of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels," September 1975. The staff evaluated and accepted this report. This report presents three acceptable alternatives, and ABB-CE's position is that a particular alternative will be specified based upon various factors

such as the configuration or the capabilities of the fabrication facility. This approach should provide adequate assurance that low-alloy steel weldments will not be developing cracking due to hydrogen. On this basis, DSER Open Item 5.2.3-1 is resolved.

ABB-CE complies with RG 1.71, "Welder Qualification for Areas of Limited Accessibility," except for the welder performance qualifications. ABB-CE's proposed alternative is for the welding supervisor to assign the most highly skilled welders to weld those joints that have the most limited accessibility. These joints are also subject to the inspections required by the applicable code. Such precautions should provide adequate assurance of the acceptability of the welding work performed on those joints that have limited accessibility. On this basis, DSER Open Item 5.2.3-2 is resolved.

In Amendment Q to CESSAR-DC Table 5.2-2, ABB-CE listed materials and information for RCPB components and included Inconel 690 materials in place of Inconel 600 materials. Operating experience indicates that Inconel 600 and

Inconel 182 (shielded metal arc welding filler material) are susceptible to cracking. ABB-CE considered alternate materials that were more resistant to cracking. ABB-CE will use Inconel 690 for all applications in which Inconel 600 was to be used. The staff views the Inconel 690 alloy as the preferred nickel alloy in the primary and secondary coolant loops because of its improved corrosion resistance compared to Inconel 600. The use of Inconel 690 will provide reasonable assurance of the material integrity of the components and tubing in contact with reactor coolant and most secondary water chemistries. On this basis, DSER Open Item 5.2.3-3 is resolved.

ABB-CE proposes to allow the continued reference to American Society for Testing and Materials (ASTM) A-708 for the purpose of maintaining the qualifications of older weld procedures. ASTM A-262, Practice A or E (recommended in RG 1.44), "Control of the Use of Sensitized Stainless Steel," has been used by ABB-CE since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. ABB-CE qualified weld procedures developed prior to the mid-1970s and used A-708 in their qualifications. These weld procedures are still in use. There have been no intergranular stress corrosion cracking (IGSCC) failures involving stainless steel weldments in ABB-CE's nuclear steam supply system (NSSS) units. In addition, the staff has allowed in SRP 4.5.1, paragraph III.2, ASTM A-708 as an acceptable alternative tests for ASTM A-262, Practice A or E. On this basis, DSER Open item 5.2.3-4 is resolved.

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ABB-CE, in Amendments L and N to the CESSAR-DC, submitted a revised Table 5.2-2 which lists the materials for RCPB components. The materials used for construction of components of the RCPB have been identified by specification and conform to the requirements of Section III of the ASME Code. Type 347 austenitic stainless steel is to be used in the surge line. The nuclear industry experienced considerable welding problems with this alloy in the 1950s. However, because of (1) the newer melting processes, such as argon-oxygen-decarbonization, which reduce the amount of carbon and impurities, (2) ABB-CE's several years of problem-free experience with welding this alloy in the System 80 plants, and (3) the section sizes involved in these components are not heavy, i.e., less than 2-inch thickness, there appears to be little risk in the weldability of this alloy in the present circumstances. On this basis, Open Item 5.2.3-5 is resolved.

The following recent fatigue test data indicate that the effects of the environment could significantly reduce the fatigue resistance of materials:

- K. Iida, J. Fukakura, M. Higuchi, H. Kobayashi, S. Miyazono, and M. Nakao, "Survey of Fatigue Strength Data of Nuclear Structural Materials in Japan," Abstract of DBA Committee Report, 1988. (Presented to the American Society of Mechanical Engineers, Subgroup on Fatigue Strength, on December 5, 1988, in New York City.) (Enclosure in letter, from John W. Craig of NRC to Edward Griffing of the Nuclear Management and Resources Council, Washington, DC, July 2, 1991.)
- M. Higuchi and K. Iida, "Fatigue Strength Correction Factors of Carbon and Low Alloy Steels in Oxygen-Containing High-Temperature Water," Nuclear Engineering and Design, Volume 129, 1991, pp. 293-306.
- J. B. Terrell, "Effect of Cyclic Frequency on the Fatigue Life of ASME SA-106-C Piping Steel in PWR Environments," Journal of Materials Engineering, Volume 10, Number 3, 1988, pp. 193-203.

The specific concern relates to the reactor water and temperature environment and its synergistic interactions with strain rate. The recent data indicate that the design fatigue curves in ASME Code Section III Class 1 requirements may not be as conservative as originally intended.

The design of System 80+ components will address the potential influence of environmental effects on the fatigue life of materials over the 60-year design life. The issue of environmental effects on fatigue is currently

under consideration by a special steering committee for cyclic life and environmental effects in nuclear applications of the pressure vessel research council (PVRC). These activities were initiated in response to requests from the ASME Boiler and Pressure Vessel (B&PV) Code Committee and the Board on Nuclear Codes and Standards (BNCS). The charter of the PVRC Steering Committee is to provide guidance and direction related to determining the effects of light-water-reactor (LWR) service environments on the cyclic life properties of applicable materials. The steering committee is also evaluating application methodologies that include these effects in the fatigue analysis process.

Preliminary recommendations were made to the BNCS in September 1992. The initial findings reported to BNCS were that the current fatigue S/N curves should be appropriate for PWR environments. There was not a complete agreement of the steering committee on this position and the issue is unresolved. ABB-CE will continue to monitor the industry activities on the fatigue curves and fatigue analysis methodology.

System 80+ components will be designed to the ASME B&PV Code requirements. ABB-CE addressed the issue of environmental fatigue of Class 1 components in Amendment R to Section 3.9.1 of CESSAR-DC and the staff's evaluation is provided in Section 3.12.5.7 of this report. On this basis, DSER Open Item 5.2.3-6 is resolved.

In response to Q281.46, ABB-CE stated that the CVCS maintains the reactor coolant chemistry within the limits defined in Electric Power Research Institute (EPRI) guidelines (EPRI, "PWR Primary Water Chemistry Guidelines: Revision 2," Report NP-7077, November 1990). In addition, in a letter of May 8, 1992 (letter number LD-92-064), ABB-CE indicated that the reactor coolant chemistry is consistent with EPRI Utility Requirements Document (URD) (EPRI, "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, ALWR Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991). However, the water chemistry stated in CESSAR-DC Section 5.2.3.4.1.2.1 was not consistent with EPRI NP-7077 and EPRI URD.

In a letter of November 24, 1992, ABB-CE identified the chemistry limits in CESSAR-DC Section 5.2.3.4.1.2.1 as applying to cleaning water used on NSSS components fabricated with austenitic stainless steel and, accordingly, would not be consistent with reactor coolant water chemistry limits. These cleaning water chemistry limits provide contamination control during component fabrication, shipment, and storage as

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required by RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants." Also, to prevent halide-induced intergranular corrosion, which would occur with the presence of significant quantities of oxygen, flushing water is inhibited by additions of hydrazine. The staff finds that the cleaning and flushing water does not have to conform to the chemistry limits of reactor coolant water.

In addition, EPRI URD indicates that the guidelines in EPRI NP-7077 should be supplemented as follows for the makeup water storage tank:

Conductivity at 25°C	0.2 μ S/cm
Oxygen	0.100 ppm (max)
Chloride	0.15 ppm (max)
Fluoride	0.15 ppm (max)
Suspended solids*	1.0 ppm (max)

*Solids concentration is determined by filtration through a filter having a 0.45-micron pore size.

In Amendment J, ABB-CE revised the primary water chemistry in CESSAR-DC Table 9.2.3-1 to meet or exceed the guidelines of EPRI URD. On this basis, DSER Open Item 5.2.3-7 is resolved.

ABB-CE initially proposed to fabricate the pressurizer surge line and the RCS valves from cast austenitic stainless steel (SA351 CF8M). Thermal aging of cast austenitic stainless steel at the reactor temperature reduces the fracture toughness of the materials (U.S. Nuclear Regulatory Commission, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, June 1991). ABB-CE needed to address the effect of thermal embrittlement for the 60-year plant design life.

In addition, it is difficult to inspect cast austenitic stainless steel using ultrasonic techniques. As discussed in Sections 5.2.4 and 6.6 of this report, 10 CFR 50.55a(g) requires ASME Code Class 1, 2, and 3 components to be designed to enable the performance of inservice examination in accordance with ASME Code Section XI. Further, 10 CFR 50.55a(g) requires the performance of a preservice inspection in accordance with ASME Code Section XI. This regulation has no relief provisions for preservice inspection. A material that is difficult to

inspect may not be able to conform to 10 CFR 50.55a(g).

Because of the thermal aging and inspection concerns, ABB-CE considered such alternatives to cast materials as wrought materials.

As a result, ABB-CE modified the CESSAR-DC to specify that the surge line will be fabricated from wrought stainless steel product forms. Therefore, thermal aging and the difficulties associated with ultrasonic inspection of cast stainless steel will not apply to the surge line.

In addition, ABB-CE considered alternative materials to cast stainless steels for other components. Where cast stainless steel was determined to be the best material selection for a specific application, such as large-diameter valve bodies, ABB-CE provided, in Amendment Q to CESSAR-DC, additional controls. These controls involved further restriction of the permissible ferrite content of the casting to minimize the effects of thermal aging. ABB-CE also considered the component service temperature for the 60-year design life since at lower operating temperatures, thermal aging is of less concern. The standard range of permissible ferrite content specified for stainless steel castings is from 8 percent to 30 percent.

In Amendment Q to the CESSAR-DC, ABB-CE limited the ferrite content of austenitic stainless steel castings to a maximum of 20 percent. The maximum ferrite content limit of 20 percent will reduce the effects of thermal aging in stainless steel castings when service temperatures are at the higher temperatures of the normal operating range (e.g., > 500 °F (260 °C)). This should provide assurance that the cast stainless steel components will perform their intended function for the 60-year life of the plant. On this basis, DSER Open Item 5.2.3-8 is resolved.

The ferrite content limits for austenitic stainless steel castings and weld metal given in the CESSAR-DC are broader than those in industry guidelines (EPRI URD) and staff guidance (NUREG-CR/4513). In Amendment Q to the CESSAR-DC, ABB-CE limited the ferrite content of austenitic stainless steel castings with normal operating temperatures above 500 °F (260 °C) to a maximum of 20 percent. In Amendment L to the CESSAR-DC, ABB-CE modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes the lower ferrite content limits for austenitic stainless steel castings and weld metal will provide reasonable assurance that components of these materials maintain adequate fracture toughness for their

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60-year life. On this basis, DSER Open Item 5.2.3-9 is resolved.

ABB-CE's use of lubricants within the RCPB is limited to small amounts for bolting on the reactor and reactor coolant pump internals and as lubrication for closure studs. These lubricants are either graphite or nickel in either an alcohol, silicon, or petroleum oil base with tightly controlled limits on halogens and sulfur. No molybdenum disulfide lubricants are used. Accordingly, reasonable assurance of fastener integrity in contact with reactor coolant is provided. On this basis, DSER Open Item 5.2.3-10 is resolved.

In Chapter 1, paragraph 5.3.1.1.5 of the URD, EPRI states that although the stated controls on grinding on stainless steel may be desirable for use on PWR components, they are not required for avoiding IGSCC. ABB-CE intends to avoid fabrication processes which would severely cold work the surface of austenitic stainless steel components. Grinding is done with resin-bonded or rubber bonded, aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic alloys. This will offer reasonable assurance of stainless steel component integrity, and that stainless steel components in contact with reactor coolant are not likely to crack. On this basis, DSER Open Item 5.2.3-11 is resolved.

ABB-CE restricts the use of SA 540 Grade B23 or B24 bolts in the RCPB to Class 3 material which has a maximum ultimate tensile strength of 1,172 MPa (170 ksi). The use of this class of material complies with the requirements of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." Laboratory and field experience with these materials has indicated resistance to stress corrosion cracking under anticipated service conditions. Accordingly, these materials (with a restricted ultimate tensile strength and the periodic ISI requirements of the code) should provide adequate assurance of the structural integrity of the RCPB for the life of the plant. On this basis, DSER Open Item 5.2.3-12 is resolved.

Carbon steel materials may be susceptible to the mechanism of dynamic strain aging which reduces the material fracture properties (C.W. Marschall, M.P. Landow, and G.M. Wilkowski, "Effect of Dynamic Strain Aging on Fracture Resistance of Carbon Steels Operating at Light-Water-Reactor Temperatures," ASTM STP 1074, ASTM, Philadelphia, PA, 1990, pp. 339-360.) ABB-CE is applying the leak-before-break concept to the RCPB. When established NRC procedures for leak-before-break of RCPB components are applied, adequate material

fracture toughness properties exist so that the effects of dynamic strain aging will not be of concern. On this basis, DSER Open Item 5.2.3-13 is resolved.

In CESSAR-DC Sections 5.2.3.4.1.1.1 and 3.6.3.5, ABB-CE proposes using Types 304 and 316 austenitic stainless steel in the RCPB. These materials are susceptible to IGSCC when the oxygen content of the reactor coolant exceeds 0.010 ppm at temperatures above 93 °C (200 °F) during normal operations. During startup and operation of the ABB-CE NSSS, these temperature and chemical conditions are avoided through water chemistry controls specified in the CESSAR-DC. ABB-CE has taken alternative mitigating approaches as allowed in RG 1.44, thus providing reasonable assurance of the integrity of austenitic stainless steel components in contact with reactor coolant. On this basis, DSER Open Item 5.2.3-14 is resolved.

ABB-CE referenced RG 1.2, "Thermal Shock to Reactor Pressure Vessels," in the CESSAR-DC. The NRC has withdrawn RG 1.2 (56 FR 36175; July 31, 1991). In Amendment N, ABB-CE modified the CESSAR-DC to remove this reference from the CESSAR-DC. On this basis, DSER Open Item 5.2.3-15 is resolved.

In CESSAR-DC Section 5.2.3.4.1.1.1, ABB-CE limits the carbon content of austenitic stainless steel to 0.065 percent compared to the usual specification limit of 0.08 percent and imposed controls on welding, fabrication, and water chemistry during startup and operation to mitigate the occurrence of IGSCC in these stainless steels. These mitigating actions have proved successful in operating reactors for more than 15 years and offer reasonable assurance that IGSCC can be controlled and, therefore, maintain the integrity of these components. On this basis, DSER Open Item 5.2.3-16 is resolved.

The staff concludes that the plant design is acceptable and meets the requirements of GDC 1; 4, "Environmental and Dynamic Effects Design Basis;" 14, "Reactor Coolant Pressure Boundary;" 30, "Quality of Reactor Coolant Pressure Boundary;" and 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50; the requirements of Appendices B and G of 10 CFR Part 50; and the requirements of 10 CFR 50.55a.

The materials used for construction of components of the RCPB have been identified by specification and are in conformance with the requirements of Section III of the ASME Code. Compliance with these code provisions for material specifications conforms to the quality standards of GDC 1 and 30, and 10 CFR 50.55a.

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The materials of construction of the RCPB exposed to the reactor coolant are identified and all of the materials are compatible with the primary coolant water, which will be chemically controlled in accordance with appropriate water chemistry specifications in the CESSAR-DC. This compatibility has been proved by extensive testing and satisfactory performance. The use of austenitic stainless steel conforms to the recommendations of RG 1.44, "Control of Sensitized Stainless Steel."

General corrosion of all materials in contact with the reactor coolant is negligible and, accordingly, general corrosion is not a concern. For these materials, conservative corrosion allowances have been provided for all exposed surfaces in accordance with the ASME Code, Section III. The evidence of compatibility with the coolant and compliance with the code provisions satisfy the requirements of GDC 4 relative to compatibility of components with environmental conditions.

In CESSAR-DC Section 5.2.3.2.2, ABB-CE states that metallic materials in contact with reactor coolant are restricted in cobalt content to as low a level as practical for all stainless steel or nickel alloy components that have a large wetted surface area. Cobalt alloys are avoided except where no proven alternative exists. This is to address ALARA concerns regarding the activation of cobalt and the resultant radioactivity.

The materials of construction for the RCPB are compatible with the thermal insulation used in these areas and are in conformance with the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Conformance with the above recommendations fulfills the requirements of GDC 14 and 31 relative to prevention of failure of the RCPB.

The ferritic steel tubular products and the tubular products fabricated from austenitic stainless steel have been found to be acceptable by nondestructive examinations in accordance with the provisions of ASME Code, Section III. Compliance with these code requirements fulfills the quality standards of GDC 1 and 30, and 10 CFR 50.55a.

The fracture toughness tests required by the ASME Code, augmented by Appendix G to 10 CFR Part 50, offer reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RCPB. The use of Appendix G of the ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the code and the Nuclear Regulatory Commission (NRC) regulations in establishing safe operating procedures, give

adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these code provisions and NRC regulations fulfills the requirements of GDC 31 and 10 CFR 50.55a regarding prevention of fracture of the reactor coolant pressure boundary.

The controls imposed on welding preheat temperatures for welding ferritic steels are in conformance with the recommendations of RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," or acceptable alternative approaches to the RG's guidelines are provided. These controls offer reasonable assurance that components made from low-alloy steels will not crack during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment. These controls satisfy the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a.

Electroslag welding will not be used for RCPB components; therefore, RG 1.34, "Control of Electroslag Weld Properties," does not apply.

The controls imposed on welding ferritic steels under conditions of limited accessibility are in conformance with the recommendation of RG 1.71, "Welder Qualification for Areas of Limited Accessibility," or provide acceptable alternative approaches and ensure that proper requalification of welders is required in accordance with the welding conditions. These controls also conform to the quality standards requirements of GDC 1 and 50, and 10 CFR 50.55a. The controls imposed on weld cladding of low-alloy steel components by austenitic stainless steel are in conformance with the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low Alloy Steel Components." These controls ensure that practices that could result in underclad cracking are restricted. The controls also fulfill the quality standards of GDC 1 and 30, and 10 CFR 50.55a.

The controls to avoid stress corrosion cracking in RCPB components constructed of austenitic stainless steels include, (1) not using cold-worked austenitic stainless steel to limit the yield strength, and (2) conforming to the recommendations of RG 1.44, "Control of the Use of Sensitized Stainless Steel," and RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Plants." The controls followed in accordance with these recommendations during material selection, fabrication, examination, and protection, prevent excessive yield strength, sensitization and contamination, and offer reasonable assurance that the austenitic stainless steel RCPB components will be in a metallurgical condition that minimizes susceptibility to stress corrosion cracking

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during service. These controls fulfill the requirements of GDC 4 relative to compatibility of components with environmental conditions and the requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

The controls imposed during welding of austenitic stainless steels in the RCPB conform to the recommendations of RG 1.31, "Control of Stainless Steel Welding," and RG 1.71, "Welder Qualification for Areas of Limited Accessibility," or provide acceptable alternative approaches. These controls offer reasonable assurance that welded components of austenitic stainless steel will not develop microfissures during welding and will have high structural integrity. These controls fulfill the quality standards requirements of GDCs 1 and 30, and 10 CFR 50.55a, and conform to the requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

The staff reviewed the RCPB ISI requirements stated in the CESSAR-DC to the criteria in SRP Section 5.2.4. To ensure that no deleterious defects develop during service, selected welds and weld heat-affected-zones will be inspected before plant startup and periodically throughout the life of the plant. The areas reviewed included the inspection requirements for the RCPB, accessibility of welds, examination categories and methods, inspection intervals, evaluation of examination results, system leakage and hydrostatic pressure tests, code exemptions, and relief requests, as appropriate. DSER open items were resolved as follows:

ABB-CE committed to comply with the design and access for inspection provisions for Class 1 components of the RCPB of the ASME Code, Section XI, in Amendment N to the CESSAR-DC. The applicable edition of the code (1989) gives specific requirements for accessibility to accomplish preservice and ISIs and, therefore, should offer adequate assurance of accessibility for preservice and ISIs. On this basis, DSER Open Item 5.2.4-1 is resolved.

In Amendment N to the CESSAR-DC, ABB-CE revised the CESSAR-DC to require that preservice inspection (PSI) will conform to Section XI of the same edition as the ASME Code used for construction and ISI will conform to Section XI in effect in accordance with 10 CFR 50.55a(g). On this basis, DSER Open Item 5.2.4-2 is resolved.

In response to Q252.17, ABB-CE indicated that it will submit relief requests to the NRC when Section XI PSI requirements are impractical. However, because the PSI requirements are established and can be incorporated into the component design, 10 CFR 50.55a(g) does not have provisions for relief requests for impractical PSI examination requirements. In Amendment N to the CESSAR-DC, ABB-CE revised the CESSAR-DC to require that PSI be in accordance with Section XI of the same edition as the ASME Code used for construction. On this basis, DSER Open Item 5.2.4-3 is resolved.

The COL applicant will have the overall responsibility for the completion of the PSI and ISI of the RCPB throughout the service life of the plant. However, the plant should be designed for accomplishment of preservice and ISIs, and in particular, should reflect the requirements of Appendix VIII of the 1989 addenda to the ASME Code which ABB-CE has acknowledged.

In Amendment U to the CESSAR-DC, ABB-CE modified Section 5.2.4 of the CESSAR-DC to state that systems and components are designed so that design, materials and geometry do not restrict inspections required by Section XI of the ASME Code. The basis for requests for relief is in 10 CFR 50.55a(g)(4). ABB-CE eliminated the need for relief under this paragraph, and accordingly, COL applicants should not need to request relief because materials, design, or geometry make ISI examinations and tests impractical. A review of the history of relief requests for Palo Verde, Units 1, 2 and 3, in "Commercial Nuclear Power Plant ISI Relief Request Data Base Regarding ASME Code Section XI Requirements" prepared by The Idaho National Engineering Laboratory showed that no requests for relief have been granted to date for material or design limitations. Two were granted for reason of limited access (removal of insulation) which, for a plant designed under the 1989 edition of the Code, would not be an acceptable reason for granting relief because this particular edition of the code requires that access be provided to perform PSI and ISI examinations. The staff views the Palo Verde experience as evidence that ABB-CE has maintained a philosophy of designing its units so that code-required inspections can be performed. On this basis, DSER Open Item 5.2.4-4 is resolved.

ASME Section XI states that PSI should be conducted with equipment and techniques equivalent to those that are expected to be used for subsequent ISI. The PSI provides the baseline information for reference in subsequent ISI. For example, if the ISI of piping weld is expected to be performed with ultrasonic techniques, the PSI should also be based on ultrasonic techniques. In Amendment L to the CESSAR-DC, ABB-CE specified

that PSI and subsequent ISI will be conducted with equivalent equipment and technologies. On this basis, DSER Open Item 5.2.4-5 is resolved.

The ASME has published ultrasonic testing standards in Section XI, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The NRC has approved the 1989 edition of the ASME Code, Section XI which includes Appendix VII. The applicable code edition for the CESSAR-DC is the 1989 edition. In Amendment N, ABB-CE revised the CESSAR-DC to require that the Section XI requirements are to be augmented with the requirements in Appendix VIII. On this basis, DSER Open Item 5.2.4-6 is resolved.

ASME Class 1, 2, and 3 carbon and low-alloy steel piping items that are susceptible to wall thinning as a result of the single phase (water) erosion-corrosion phenomenon will be subject to examination in accordance with Subsection IWH of ASME Code Section XI.

As noted in CESSAR-DC Sections 5.2.4 and 6.6.1, although ASME B&PV Code, Section XI, Subsection IWH has not yet been formally approved for use by the nuclear industry, the COL applicant will review Subsection IWH after NRC approval, and ensure that the System 80+ design PSI and ISI programs comply. Where the design will not practically accommodate full adherence to IWH, the COL applicant may propose alternatives as allowed in 10 CFR 50.55a(a)(3)(i) and (ii). This is part of COL Action Item 5-3 as noted in CESSAR-DC Table 1.10-1. On this basis, DSER Open Item 5.2.4-7 is resolved.

Table IWB-2500-1 in ASME Code Section XI requires the examination of Class 1 piping welds, with a Section III fatigue cumulative usage factor (CUF) exceeding 0.4, at every inspection interval. ABB-CE has confirmed that the value of CUF to be used will correspond to a 60-year plant design life for Class 1 piping welds. On this basis, DSER Open Item 5.2.4-8 is resolved.

The COL applicant should submit its PSI and ISI program plans for staff review and approval. The COL applicant should verify that its PSI and ISI programs will incorporate the requirements of Appendices VII and VIII and Subsection IWH of ASME Section XI. This is COL Action Item 5.2.4-1.

In the CESSAR-DC, ABB-CE has states that its inspection program will comply with the rules published in 10 CFR 50.55a and Section XI of the ASME Code. The

design of the RCS incorporates provisions for access for ISI in accordance with Section XI of the ASME B&PV Code, 1989 edition. Suitable equipment has been developed to facilitate the remote inspection of those areas of the RCPB that are not readily accessible to inspection personnel. The ISI program will consist of a PSI plan and an ISI plan. Periodic examinations, and leakage and hydrostatic testing of pressure-retaining components of the RCPB, in accordance with the requirements of Section XI of the ASME Code and 10 CFR Part 50, will offer reasonable assurance that evidence of structural degradation or loss of leaktight integrity occurring during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the preservice and inservice examinations required by the ASME Code and by 10 CFR Part 50 constitutes an acceptable basis for conforming to inspection requirements of GDC 32, "Inspection of Reactor Coolant Pressure Boundary," and 10 CFR 50.55a.

The staff concludes that the COL applicant should be able to conduct a preservice and an ISI and testing program with few if any requests for relief. Such a program by the COL applicant should conform to GDC 32 and 10 CFR 50.55a. This conclusion is based on ABB-CE complying with the requirements of the ASME B&PV Code, Section XI, as reviewed by the staff.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. The RCPB leakage detection systems are designed to comply with this requirement. The staff reviewed the RCPB leakage detection systems in accordance with SRP Section 5.2.5 (NUREG-0800) and RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

RG 1.45 states that the source of reactor coolant leakage be identifiable to the extent practical. Specifically, Position C.1 of RG 1.45 states that the leakage to the primary reactor containment from identified sources should be collected or isolated so that the flow rates are monitored separately from unidentified leakage, and the total flow rates can be established and monitored. Position C.2 states that the leakage to the primary containment from unidentified sources should be collected and the flow rate monitored with an accuracy of 3.8 L/min (1 gpm) or better. Position C.3 states that at least three separate detection methods should be identified. Position

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C.4 states that provisions should be made to monitor systems connected to the RCPB for signs of intersystem leakage. Position C.5 states that the sensitivity and response time of each leakage detection system used for unidentified leakage should be adequate to detect a leakage rate, or its equivalent, of 3.8 L/min (1 gpm) in less than 1 hour. Position C.6 states that the leakage detection systems should be capable of performing their functions following seismic events that do not require plant shutdown. Position C.7 states that indicators and alarms for each leakage detection system should be installed in the main control room. Position C.8 states that the leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operation.

In CESSAR-DC Section 5.2.5, ABB-CE states that the leakage detection systems are designed to meet the guidance delineated in RG 1.45. Leakage is classified into two types: identified and unidentified. The leakage from such components as valves located on the RCS, reactor coolant pump seals, valves inside the containment, and reactor vessel seals is considered identified leakage, and is monitored, limited, and separated from other leakage (unidentified) by directing it to tanks or sumps. ABB-CE discusses sources, disposition, and indication of identified leakage in CESSAR-DC Section 5.2.1.2. The staff reviewed this information and concludes that it complies with Position C.1 of RG 1.45.

In CESSAR-DC Section 5.2.5.1.1, ABB-CE gives four methods for detecting unidentified leakage, namely: (1) inventory method, (2) containment air particulate and gaseous radioactivity monitoring, (3) sump level and flow method, and (4) acoustic leakage monitoring system. The system for detecting unidentified leakage is designed to detect a minimum of 3.8 L/min (1 gpm) in 1 hour or less. Therefore, ABB-CE has complied with Positions C.2, C.3, and C.5 of RG 1.45.

Intersystem leakage is detected by an increase in the interfacing system level, radioactivity, temperature, or pressure. Specifically, the intersystem leakage for the following areas are monitored: SIS, pressurizer relief valve, shutdown cooling system, component cooling water system, and SG leakage. Therefore, ABB-CE has complied with Position C.4 of RG 1.45.

In a response to the NRC request for additional information (RAI) Q410.101, ABB-CE committed to design the containment atmosphere monitor to seismic Category I and to revise CESSAR-DC Table 11.5-3 accordingly. This was designated as DSER Confirmatory Item 5.2.5-1. The staff has confirmed in Amendment J that the containment atmosphere monitor is designed to seismic

Category I. On this basis, DSER Confirmatory Item 5.2.5-1 is resolved. Furthermore, the containment atmosphere monitors are located inside the containment, a flood- and tornado-protected structure, thus conforming to the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," and the guideline Positions C.1 and C.2 of RG 1.29, "Seismic Design Classification," regarding the seismic design of the safety-related and non-safety-related portions of the system, respectively, and Position C.6 of RG 1.45.

In CESSAR-DC Section 5.2.5.2, ABB-CE notes that indicators and alarms for safety valve leakage and RCS leakage are installed in the control room. In CESSAR-DC Section 5.2.5.2.4, ABB-CE describes procedures for converting leakage detection to a common leakage equivalent. On the basis of this information, the staff concludes that the design of the leakage detection system conforms to Position C.7 of RG 1.45. In CESSAR-DC Section 5.2.5.6, ABB-CE describes testing and calibration of the leakage detection systems, demonstrating compliance to the guidelines of Position C.8 of RG 1.45.

In CESSAR-DC Section 5.2.5.3, ABB-CE states that the maximum allowable total identified leakage will be given in the TS. However, in Position C.9 of RG 1.45, the staff states that the TS should include the limiting conditions for both identified and unidentified leakage and should address the availability of various types of instruments. This was designated as DSER Open Item 5.2.5-1. In Amendment N, ABB-CE revised Section 5.2.5.3 to include both identified and unidentified leakage in the TS. TS Limiting Condition for Operation (LCO) 3.4.12 and LCO 3.4.14 include this information. On this basis, DSER Open Item 5.2.5-1 is resolved.

The RCPB leakage detection systems are diverse and give reasonable assurance that primary system leakage (both identified and unidentified) will be detected. Therefore, the systems conform to the requirements of GDC 2 and 30 with respect to protection against natural phenomena and provisions for RCPB leak detection and identification, and the guidelines of RG 1.29, Positions C.1 and C.2, and RG 1.45, Positions C.1 through C.9, with respect to the seismic classification and RCPB leakage detection systems. The detection of leakage through the RCPB is in accordance with the acceptance criteria of SRP 5.2.5 and, therefore, is acceptable.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

The staff has reviewed the reactor vessel materials in accordance with SRP Section 5.3.2. The areas of re-

view included material specifications, special processes used for manufacture and fabrication of components, special methods of nondestructive examination, special controls and special processes used for ferritic steels and austenitic stainless steels, fracture toughness, material surveillance and reactor vessel fasteners. The staff's DSER open items were resolved as follows:

When the copper content of the reactor vessel beltline material is reduced, the susceptibility of the material to embrittlement by neutron irradiation becomes dominated by other chemical elements, such as nickel and phosphorus. ABB-CE has allowed the nickel content of the reactor vessel beltline forging to vary the full range (0.4 percent to 1.00 percent) permitted by the SA 508 Class 3 material specification. The available data indicate that the influence of nickel on susceptibility to irradiation damage is limited when other residual elements, in particular copper, are controlled to very low levels. Weld metals show slightly more sensitivity to nickel at low copper contents than base metals. Therefore, the nickel content for beltline weld metal is controlled to lower levels than the forging material.

In Amendment N to the CESSAR-DC, ABB-CE reduced the phosphorus content in the reactor vessel forging and weld metal to a maximum of 0.012 percent. The possible effect of phosphorus on predicted shifts in low copper reactor pressure vessel (RPV) beltline materials was evaluated by Oak Ridge National Laboratory (ORNL) and the NRC using the PR-EDB. Only a marginal correlation was observed between phosphorus content and predicted shift for low copper materials. The conclusion that there was any significant correlation was judged to be weak. Even in those cases in which the data suggested a possible trend with phosphorus content, the differences between the observed and predicted shifts were within the margins applied by RG 1.99, Revision 2. In addition, in the cases in which the data suggested a possible trend with phosphorus content, it was generally the high phosphorus (>0.012 percent) materials which indicated the possible trend. In view of this earlier assessment of the effects of residual elements on the irradiation response of materials, the existing controls on residual elements are concluded to be sufficient. The limits on copper, phosphorus, and other residual elements will minimize the extent of radiation damage to the RPV beltline materials. The radiation-induced shifts in reference temperatures for these materials can be predicted with reasonable accuracy and conservatism using the methodology of RG 1.99, Revision 2. On this basis, Open Item 5.3.1-1 is resolved.

The staff believes that the reactor vessel materials surveillance program plan should be based on a reasonably

conservative estimate of the temperature shift. This is because it may be technically difficult to backfit an existing surveillance program should the actual temperature shift be higher than that estimated. In the CESSAR-DC, ABB-CE calculates predicted shifts in the reference temperature for the reactor vessel materials using the methodology of RG 1.99, Revision 2. This methodology produces reasonably accurate and conservative predictions of adjusted reference temperatures for RPV beltline materials, including low copper base and weld metals with phosphorus impurities controlled to low levels.

The surveillance program is based on a reasonably conservative estimate of the temperature shift. The recommended minimum number of surveillance capsules in ASTM E185 for a reactor vessel with an EOL shift between 38 °C and 93 °C (100 °F and 200 °F) is four. The System 80+ surveillance program includes six capsules with archive materials available for at least two additional complete replacement capsules which can be installed in the reactor at any time when circumstances indicate that an additional capsule is required and when there is an available holder location. On this basis, DSER Open Item 5.3.1-2 is resolved.

Appendix H to 10 CFR Part 50 requires the surveillance program to comply with ASTM E185-82. In the CESSAR-DC, ABB-CE contends that its surveillance program is designed on the basis of Appendix H to 10 CFR Part 50 and ASTM E185-82. ASTM E185-82 has been applicable to plants designed for 40 years, i.e., 32 effective full-power years (EFPYs) at EOL. ABB-CE submitted additional information on its surveillance program in response to Q252.1.

In the CESSAR-DC, ABB-CE describes the application of six surveillance capsules, three of which were scheduled for withdrawal before the end of the 60-year design life of the plant. In CESSAR-DC Table 5.3-7, ABB-CE shows the surveillance capsule withdrawal schedule. The withdrawal schedule outlined by ABB-CE upon which the DSER was based was unacceptable. The proposed capsule withdrawal schedule in the CESSAR-DC would provide a capsule data point in 5 to 8 EFPYs. The next capsule data point would not be available until 22 to 26 EFPYs. This schedule was insufficient to establish the actual trend of embrittlement early in plant life to provide for timely mitigation.

The staff required that the schedule in ASTM E185-82 be supplemented to address the period between 40 and 60 years. The staff, in the DSER, required that ABB-CE propose a capsule withdrawal schedule beyond 32 EFPYs to demonstrate compliance with 10 CFR Part 50,

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Appendix H, to the end of the plant's design life of 60 years. The staff proposed as an option that the time interval between the last two capsule withdrawals be maintained within 32 EFPYs throughout the rest of plant design life, or at the end of the 60-year plant design life, whichever is earlier.

The staff proposed the following example: If a design certification applicant estimates that the reference temperature shift is greater than 56 °C (100 °F) but less than 111 °C (200 °F), the capsule withdrawal schedule in ASTM E185-82 would require one capsule each to be withdrawn at 3, 6, 15, and 32 EFPYs, along with certain restrictions on fluence levels. ABB-CE should follow this schedule for up to 32 EFPYs. Then, ABB-CE should propose a schedule beyond 32 EFPYs to the end of the 60-year plant design life. As an option, ABB-CE may consider maintaining the time interval between the last two capsules that will be withdrawn during the 32 EFPYs; in this case, 32 minus 15, or 17 EFPYs. So, the next capsule should be withdrawn at 49 EFPYs, i.e., 32 plus 17, or at the end of the 60-year plant design life, whichever is earlier.

In Amendment L to the CESSAR-DC, ABB-CE revised the capsule withdrawal schedule to withdraw a surveillance capsule at 6, 15, 32 and 48 (EOL) EFPYs over the 60-year design life of the reactor vessel. This revision to the withdrawal schedule is consistent with the discussion given for this open item, with the exception that a surveillance capsule will not be withdrawn at 3 EFPY. The predicted shift for the beltline forging material to be used for the System 80+ reactor vessel is predicted to have approximately an 8 °C (15 °F) shift at a fluence corresponding to 3 EFPY. The predicted shift after 6 EFPYs for the beltline forging is only 10 °C (18 °F). The material selection and requirements for the System 80+ reactor vessel beltline will ensure minimal damage from irradiation and minimal changes in properties. The beltline materials in the RPVs of this design will have tighter chemistry limits and more stringent requirements for initial properties than the latest fabricated vessels of currently operating, licensed reactor vessels. The staff agrees that there is not a sufficient difference in predicted shift over this short time span relative to the overall life of the reactor vessel to justify the need for two surveillance capsules being analyzed in the first 6 EFPYs. On this basis, DSER Open Item 5.3.1-3 is resolved.

ABB-CE has clarified its intent relating to welds within the reactor vessel beltline region. In its letter of November 24, 1993, ABB-CE expressed the intent to supply beltlines without welds if such is possible at construction time. Since there is no certainty that such a

supplier will exist in the future, ABB-CE has included a series of surveillance capsules which include welds in the test program. On this basis, DSER Open Item 5.3.1-4 is resolved.

In Amendment S to CESSAR-DC, ABB-CE modified the CESSAR-DC to specify use of a nickel-containing, high-purity lubricant on the reactor vessel studs. The use of lubricants is to be limited to small amounts for bolting on the reactor and reactor coolant pump internal components and as lubrication for closure studs. These lubricants are either graphite or nickel in either alcohol, silicon, or petroleum oil base with tightly controlled limits on halogens and sulfur. Accordingly, the integrity of fasteners used in the RCPB, with the ISI requirements of the code, is reasonably assured for the life of the reactor vessel. On this basis, DSER Open Item 5.3.1-5 is resolved.

The concern relating to the environmental effects on fatigue is discussed in Section 5.2.3 of this chapter. This is applicable to all materials.

The design of System 80+ components will address the potential influence of environmental effects on the fatigue life of materials over the 60-year design life. The issue of environmental effects on fatigue is currently under consideration by a special steering committee for cyclic life and environmental effects in nuclear applications of the PVRC. These activities were initiated based on requests from the ASME B&PV Code Committee and the BNCS. The charter of the PVRC steering committee is to provide guidance and direction related to determining the effects of LWR service environments on the cyclic life properties of applicable materials. The steering committee is also evaluating application methodologies that include these effects in the fatigue analysis process.

Preliminary recommendations were made to the BNCS in September 1992. The initial findings reported to BNCS were that the current S/N curves should be appropriate for PWR environments. There was not a complete agreement of the steering committee on this position and the issue is unresolved. ABB-CE will continue to monitor the industry activities on the fatigue curves and fatigue analysis methodology.

System 80+ components will be designed to the ASME B&PV Code requirements. ABB-CE addressed the issue of environmental fatigue of Class 1 components in Amendment R to Section 3.9.1 of CESSAR-DC and the staff's evaluation is provided in Section 3.12.5.7 of this report. On this basis, DSER Open Item 5.2.3-6 is resolved.

ABB-CE takes exception to the recommendations in Position C.2 of RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," for controls imposed on preheat temperatures for welding ferritic steels. These controls would provide reasonable assurance that components made from low-alloy steels will not crack during fabrication and minimize the possibility of subsequent hydrogen cracking in the weldment. In a letter of November 24, 1992, ABB-CE stated that its basis for taking exception to Position C.2 was Westinghouse Topical Report WCAP-8678, "Effect of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels," September 1975. The NRC evaluated and accepted this report. This report presents three acceptable alternatives, and ABB-CE's position is that a particular alternative will be specified based upon the particulars of the configuration or a particular facilities capabilities. This approach should offer adequate assurance that low-alloy steel weldments will not develop cracking due to hydrogen. On this basis, DSER Open Item 5.3.1-7 is resolved.

In the CESSAR-DC, ABB-CE limits the sulfur content of welds and forgings to a maximum of 0.015 percent.

This limit is the same as is specified for the restricted chemistry limits for SA 508 Class 3 forgings which are used for the RPV beltline region.

Data compiled in EPRI Report NP-933, "Nuclear Pressure Vessel Steel Data Book," December 1978, indicated that this control on the level of sulfur will provide the fracture toughness required to ensure the structural integrity of the reactor vessel as specified by 10 CFR Part 50, Appendix G.

A heat treatment applied to a typical forging is as follows:

Austenitizing:	This temperature is established by the melter or manufacturer of the forging or plate.	840-895 °C (1,540-1,640 °F) for 4 hours followed by water quenching.
Tempering:	The temperature and time are specified by the ASME specification.	659 °C (1,200 °F) minimum for 4 hours.
Post-Weld Heat Treatment:	Temperature and time specified by the ASME Code.	607 °C ± 14 °C (1125 °F ± 25 °F) for up to 40 hours.

The specified chemical limits, heat treatment, and post-weld heat treatment cycle should provide adequate strength and toughness for the nuclear RPV design forgings and weldments for the 60-year life of the vessel. On this basis, DSER Open Item 5.3.1-8 is resolved.

Appendix G of 10 CFR Part 50 requires reactor beltline materials to have a Charpy upper-shelf energy (USE) of no less than 102 J (75 ft.lbf) initially and must maintain a USE throughout the life of the vessel of no less than 68 J (50 ft-lb), unless lower values provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. In the CESSAR-DC, ABB-CE requires the beltline base and weld metal to

have an unirradiated minimum USE of 102 J (75 ft.lbf). The 60-year plant design life (48 EFPY) will result in an EOL fluence on the reactor vessel of 6.0×10^{19} n/cm². In accordance with the guidelines of RG 1.99, Revision 2, this fluence would result in a maximum of a 29-percent decrease in USE or a minimum of 72 J (53 ft.lbf). ABB-CE will satisfy the initial USE requirements and will meet the EOL requirement by limiting the amounts of copper and fluence to the reactor vessel beltline materials. On this basis, DSER Open Item 5.3.1-9 is resolved.

As discussed in Section 5.2.3 of this chapter, ABB-CE proposes to allow the continued reference to ASTM A-708 for the purpose of maintaining the qualifications

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of older weld procedures. ASTM A-262, Practice A or E (recommended in RG 1.44) has been used by ABB-CE since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. ABB-CE qualified weld procedures developed prior to the mid-1970s and used A-708 in their qualifications. These weld procedures are still in use. There have been no IGSCC failures involving stainless steel weldments in ABB-CE's NSSS units. In addition, the staff has allowed in SRP 4.5.1, paragraph III.2, A-708 is an acceptable alternative test for A-262, Practice A or E. On this basis, DSER Open Item 5.3.1-10 is resolved.

As discussed in Section 5.2.3 of this chapter, the ferrite content limits for austenitic stainless steel castings and weld metal in the CESSAR-DC are broader than those in industry guidelines and staff guidance. In Amendment Q to the CESSAR-DC, ABB-CE limited the ferrite content of austenitic stainless steel castings with normal operating temperatures above 500 °F (260 °C) to a maximum of 20 percent. In Amendment L to the CESSAR-DC, ABB-CE modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes the lower ferrite content limits for austenitic stainless steel castings and weld metal will offer reasonable assurance that components of these materials will maintain adequate fracture toughness for their 60-year life. On this basis, DSER Open Item 5.3.1-11 is resolved.

The staff concludes that the reactor vessel materials are acceptable and comply with the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A of 10 CFR Part 50; the material testing and monitoring requirements of Appendices B, G, and H of 10 CFR Part 50; and the requirements of 10 CFR 50.55a. This conclusion is based on the following:

- (1) The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with Section III of the ASME Boiler & Pressure Vessel Code. Special requirements with regard to control of the residual elements have been identified and are considered acceptable. Compliance with the cited code provisions for material specifications conforms to the quality standards of GDC 1 and 30, and 10 CFR 50.55a.
- (2) Special processes used for the manufacture or fabrication of the reactor vessel and its appurtenances have been identified, and appropriate data reports on each process as required by Section III of the ASME Code have been submitted by ABB-CE. Since ABB-CE has certified that it will comply with the requirements of Section III of the ASME Code, the special processes used are considered acceptable. Compliance with these code provisions conforms to the quality standards of GDC 1 and 30, and 10 CFR 50.55a.
- (3) Special methods used for nondestructive examination of the reactor vessel and its appurtenances have been identified and have been found equivalent or superior to the techniques described in Section V of the ASME B&PV Code. ABB-CE has committed to Appendixes VII and VIII of Section XI of the code which describe qualification of nondestructive examination personnel for ultrasonic examination and performance demonstrations for ultrasonic examination systems. The special methods of nondestructive examination are deemed acceptable. This acceptability, based on the code provisions, complies with the requirements of GDC 1 and 30, and 10 CFR 50.55a.
- (4) Special controls and special welding processes used for welding the reactor vessel and its appurtenances have been identified and found to be qualified in accordance with the requirements of ASME Code Sections III and IX. Qualification in accordance with the code provisions complies with the requirements of GDC 1 and 30, and 10 CFR 50.55a concerning quality standards.
- (5) When welding ferritic steel components as identified above, code controls are supplemented by conformance with the recommendations of RGs as follows:
 - The controls imposed on welding preheat temperatures are in conformance with recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low Alloy Steel," or alternative approaches which are acceptable to the staff. These controls give reasonable assurance that components made from low-alloy steels will not crack during fabrication and the potential for subsequent cracking is minimized. These controls also satisfy the quality standards of GDC 1 and 30, and 10 CFR 50.55a.
 - The controls imposed on electroslag welding of ferritic steels are not necessary because electroslag welding will not be used for RCPB components.
 - The controls imposed during weld cladding of ferritic steel components are in conformance with the recommendations of RG 1.43, "Control of

Stainless Steel Weld Cladding of Low Alloy Steel Components," because the process used provides reasonable assurance that the undercladding will not crack during the weld cladding process. These controls satisfy the quality standards of GDC 1 and 30, and 10 CFR 50.55a.

- (6) When welding austenitic stainless steel components, code controls are supplemented by conformance with the recommendations of the RGs as follows:

The controls imposed on delta ferrite in austenitic stainless steel welds are in conformance with the recommendations of RG 1.31, "Control of Stainless Steel Welding," because the controls used provide reasonable assurance that the welds will not contain micro-cracks. These controls also satisfy the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a, and the requirements of GDC 14 regarding fabrication to prevent rapid propagating failure of the RCPB.

- (7) The controls (during all stages of welding) to avoid contamination and sensitization that could cause stress corrosion cracking in austenitic stainless steels conform with the recommendations of RGs as follows:

- The controls to avoid contamination and sensitization of austenitic stainless steel are in conformance with the recommendations of RG 1.44, "Control of the Use of Sensitized Stainless Steel," because the controls used offer assurance that welded components will not be contaminated nor excessively sensitized preceding or during the welding process. These controls satisfy the quality standards requirement of GDC 1 and 30, and 10 CFR 50.55a, and the GDC 4 requirement relative to material compatibility.
- The controls regarding onsite cleaning and cleanliness control of austenitic stainless steel are in conformance with the recommendation of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," because the controls used offer assurance that austenitic stainless steel components will be properly cleaned on site. The controls satisfy Appendix B of 10 CFR Part 50 regarding controls for onsite cleaning of materials and components.

- (8) Fracture toughness of the reactor vessel and its appurtenances is controlled by conformance with Appendix G to 10 CFR Part 50, which specifies ASME Code provisions and supplementary requirements of Appendix G to 10 CFR Part 50. The fracture toughness tests required by the ASME Code, and by Appendix G to 10 CFR Part 50, provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G of the code as a guide in establishing safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the code and NRC regulations, will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with the provisions of Appendix G, 10 CFR Part 50, satisfies the requirements of GDC 14 and 31, and 10 CFR 50.55a regarding prevention of fracture of the RCPB.

- (9) Changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation have been assessed properly, and adequate safety margins against the possibility of vessel failure are provided as the material surveillance requirements (SRs) of ASTM E185 and Appendix H, 10 CFR Part 50, are met. Compliance with these documents ensures that the surveillance program serves as an acceptable basis for monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and conforms to the requirements of GDC 32 regarding an appropriate material surveillance program for the reactor vessel.

- (10) Integrity of the reactor vessel studs and fasteners is ensured by conformance with the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." Compliance with these recommendations satisfies the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a; the prevention of fracture of the RCPB requirement of GDC 31; and the requirements of Appendix G, 10 CFR Part 50, as detailed in the provisions of the ASME Code, Sections II and III.

The COL applicant should verify that the CESSAR-DC assumptions of material properties and 60-year neutron fluence relating to Appendices G and H of 10 CFR Part

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50 are met based on actual plant-specific values. This is COL Action Item 5.3.1-1.

5.3.2 Pressure-Temperature Limits

The staff has reviewed the pressure-temperature limits on the RCPB in accordance with SRP Section 5.3.2. The areas reviewed were the applicable regulations, codes, basis documents, technical bases, and the pressure-temperature requirements. The staff's open items in the DSER were resolved as follows:

ABB-CE calculated predicted shifts in the reference temperature for the reactor vessel materials using the methodology of RG 1.99, Revision 2. This approach is acceptable for domestically produced steels; however, steels from other sources could have different characteristic responses to radiation embrittlement, particularly steels with high phosphorous content. Hence, the methodology of RG 1.99, Revision 2 may not be appropriate. RG 1.99, Revision 2 provides reasonably accurate and conservative predictions of adjusted reference temperatures for RPV beltline materials that are produced domestically. COL applicants that utilize foreign steel for beltline materials may need to estimate neutron irradiation embrittlement with a methodology different from that specified in RG 1.99, Revision 2. Since ABB-CE has controlled the amount of residual elements (copper, phosphorous, sulfur), it is anticipated that embrittlement will proceed at a low rate. However, the test results from the surveillance program should provide data to accurately determine the rate of embrittlement in both domestic and foreign produced materials.

As discussed in the resolution of DSER Open Item 5.3.1-1, the phosphorus content in the reactor vessel forgings and weld metal is controlled to a maximum of 0.012 percent. The possible effect of phosphorus on predicted shifts in low copper RPV beltline materials was evaluated by ORNL and the NRC using the Embrittlement Database (EDB). Only a marginal correlation was observed between phosphorus content and predicted shift for low copper materials. The conclusion that there was any significant correlation was judged to be weak. Even in those cases in which the data suggested a possible trend with phosphorus content, the differences between the observed and predicted shifts were within the margins applied by RG 1.99, Revision 2. In addition, in the cases in which the data suggested a possible trend with phosphorus, it was generally the high phosphorus (>0.012 percent) materials which indicated the possible trend. In view of this earlier assessment of the effects of impure elements on the irradiation response of materials, the existing controls on residual elements are concluded to be sufficient. The

limits on copper, phosphorus, and other residual elements will minimize the extent of radiation damage to the RPV beltline materials. The radiation-induced shifts in reference temperatures for these materials can be predicted with reasonable accuracy and conservatism using the methodology of RG 1.99, Revision 2. On this basis, DSER Open Item 5.3.2-1 is resolved.

In Amendment N to the CESSAR-DC, ABB-CE modified CESSAR-DC to conform with the recommendation of Section 3 of the ASME Code, paragraph G.2400(a). The preservice hydrostatic test is to be performed at an isothermal temperature no lower than $RT_{NDT} (-12\text{ }^{\circ}\text{C}) + 33\text{ }^{\circ}\text{C} = 21\text{ }^{\circ}\text{C}$ ($10\text{ }^{\circ}\text{F} + 60\text{ }^{\circ}\text{F} = 70\text{ }^{\circ}\text{F}$). On this basis, DSER Open Item 5.3.2-2 is resolved.

The responses to Q252.1, Q252.5, and Q252.13 stated that the margin is $19\text{ }^{\circ}\text{C}$ ($34\text{ }^{\circ}\text{F}$) for the base metal. However, ABB-CE assumed the margin to be $28\text{ }^{\circ}\text{C}$ ($50\text{ }^{\circ}\text{F}$). In response to Q252.5, ABB-CE submitted the equation used to calculate the margin. A factor of 2 was missing from this equation when compared to the guidance given in RG 1.99, Revision 2. ABB-CE evaluated this discrepancy and determined that the response in RAI Q 252.5 was incorrect.

Based on Position 1.1 of RG 1.99, Revision 2, the margin to be added to obtain values of adjusted reference temperature (ART) can be calculated based on $\sigma_I = 0\text{ }^{\circ}\text{C}$ ($0\text{ }^{\circ}\text{F}$), and $\sigma\delta = 15.5\text{ }^{\circ}\text{C}$ ($28\text{ }^{\circ}\text{F}$) for welds and $9.5\text{ }^{\circ}\text{C}$ ($17\text{ }^{\circ}\text{F}$) for base metal.

The resultant margin according to the RG 1.99 equation

$$M = 2 [(\sigma_I)^2 + (\sigma\delta)^2]^{1/2}$$

is $19\text{ }^{\circ}\text{C}$ ($34\text{ }^{\circ}\text{F}$) for base metal and $31\text{ }^{\circ}\text{C}$ ($56\text{ }^{\circ}\text{F}$) for welds.

As stated in Section 5.3.2.1.1 of the CESSAR-DC, a margin of $27.8\text{ }^{\circ}\text{C}$ ($50\text{ }^{\circ}\text{F}$) was used for base metal (ring forgings) to provide an added degree of conservatism based upon RG 1.99, Revision 2. However, since the issuance of RG 1.99, Revision 2, the staff has performed a statistical analysis of all plate surveillance data that have been received from licensees. The analysis of the surveillance data indicates that the standard deviation for the increase in reference temperature should be $14.4\text{ }^{\circ}\text{C}$ ($26\text{ }^{\circ}\text{F}$), resulting in a margin of $28.9\text{ }^{\circ}\text{C}$ ($52\text{ }^{\circ}\text{F}$). The staff has raised this issue in the PTS review of other plants, such as Indian Point 3 and Beaver Valley 1. The staff believes that using a margin of $27.8\text{ }^{\circ}\text{C}$ ($50\text{ }^{\circ}\text{F}$) rather than $28.9\text{ }^{\circ}\text{C}$ ($52\text{ }^{\circ}\text{F}$) is adequate because ABB-CE has controlled the amount of residual elements (copper, nickel and phosphorous), which will result in less

embrittlement than the vessels currently operating in the U.S. On this basis, DSER Open Item 5.3.2-3 is resolved.

The responses to Q252.1 and Q252.5 stated that the initial RT_{NDT} for the beltline material is -29°C (-20°F). However, the response to Q252.13 stated that the initial RT_{NDT} is conservatively assumed to be -12°C (10°F) for PTS calculations. In Amendment N to the CESSAR-DC, ABB-CE modified the CESSAR-DC and specified an initial RT_{NDT} value of -12°C ($+10^{\circ}\text{F}$) for the beltline material in all applicable calculations. This approach is satisfactory to the staff. On this basis, DSER Open Item 5.3.2-4 is resolved.

The pressure-temperature limits imposed on the RCS for operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure are in conformance with the fracture toughness criteria of Appendix G to 10 CFR Part 50 and Section III, including Appendix G, of the ASME Code. The change in fracture toughness requirements of the pressure vessel during operation will be determined by Appendix H to 10 CFR Part 50. The use of operating limits, based upon the criteria defined in SRP Section 5.3.2, offers reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a and GDC 1, 14, 31, and 32 (Appendix A to 10 CFR Part 50).

The COL applicant should submit plant-specific material fracture toughness data and the resulting pressure-temperature curves for staff review and approval. This is COL Action Item 5.3.2-1.

5.3.3 Reactor Vessel Integrity

The staff concludes that the structural integrity of the reactor vessel is acceptable and conforms to the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; the requirements of Appendices B, G, and H of 10 CFR Part 50; and the requirements of 10 CFR 50.55a. This conclusion is based on the staff's review of the CESSAR-DC conducted in accordance with the following SRP sections, and supplemented by the acceptance criteria of SRP Section 5.3.3:

- SRP Section 5.2.3, "RCPB Materials"
- SRP Section 5.2.4, "RCPB Inservice Inspection and Testing"
- SRP Section 5.3.1, "Reactor Vessel Materials"
- SRP Section 5.3.2, "Pressure-Temperature Limits"

The staff reviewed all factors contributing to the structural integrity of the reactor vessel and concludes there are no special considerations that make it necessary to consider potential reactor vessel failure for this design. The basis for the conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements for the CESSAR-DC will conform to applicable NRC regulations and RGs, and to the rules of the ASME B&PV Code, Section III. The stringent fracture toughness requirements of the regulations and ASME Code, Section III will be adhered to, including requirements for surveillance of vessel material properties throughout service life, in accordance with Appendix H to 10 CFR Part 50. Also, operating limitations of temperature and pressure will be established for this plant in accordance with Appendix G, "Protection Against Nonductile Failure," of ASME Code Section III, and Appendix G to 10 CFR Part 50.

The integrity of the vessel is assured because the vessel:

- will be designed and fabricated to the high standards of quality required by the ASME B&PV Code and any pertinent code cases
- will be made from materials of controlled and demonstrated high quality
- will be subjected to extensive preservice examination and testing to provide assurance that the vessel will not fail because of material or fabrication deficiencies
- will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients
- will be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions
- may be annealed to restore the material properties if this becomes necessary
- will be subjected to surveillance to account for neutron irradiation damage so that the operating conditions may be adjusted

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5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

Pump Flywheel Integrity

The staff reviewed the reactor coolant pump flywheel integrity in accordance with SRP Section 5.4.1.1. The areas of review were materials selection, fracture toughness, preservice inspection, flywheel design, overspeed test, and ISI. The staff's open items in the DSER were resolved as follows:

In response to Q252.08, ABB-CE stated that the design of the pump flywheel depends on the motor supplier. The material selected may be plate or forging. ABB-CE stated that a typical plate material is A-533, Grade B, Class 1, and a typical forging material is A-508, Class 2 or Class 3. However, the recommended materials for flywheels listed in SRP Section 5.4.1.1 are ASME SA-533-B Class 1, SA-508 Class 2, and SA-516 Grade 65. The SA-508 Class 3 material is not on this recommended list. In a letter of November 24, 1993, ABB-CE justified the use of SA-508 Class 3 material.

SA-508 Class 3 is a quenched and tempered vacuum-treated carbon and alloy steel forging material commonly used for pressure vessel construction. The material is a high-strength ductile material produced in different classes from Class 1 to Class 5 by varying the chemical composition and heat treatment temperatures. SA-508 Class 2 and SA-508 Class 3 are essentially the same material. Both materials have the same heat treatment temperatures and mechanical properties. The only differences are slight variations in the carbon content (0.27 max. versus 0.25 max.) and the alloy compositions as shown in the SA-508 material specification. Figure A-4200-1 in ASME Section XI shows the same lower bound fracture toughness curve for SA-508 Class 2 and SA-508 Class 3 materials.

ABB-CE considers SA-508 Class 3 to be an acceptable flywheel forging material and requests that it be added to SRP Section 5.4.1.1 to allow for future use of other acceptable materials.

The staff believes this material should provide adequate materials properties to ensure a 60-year life of the reactor coolant pump flywheel. On this basis, DSER Open Item 5.4.1.1-1 is resolved.

Originally, ABB-CE proposed, as an alternative to testing the fracture toughness of the actual material, to use a lower bound fracture toughness curve obtained from

testing the same type of material and adjusting the curve to provide a dynamic stress intensity factor of $49 \text{ MPa}\sqrt{\text{m}}$ ($45 \text{ ksi}\sqrt{\text{in.}}$) at the nil ductility temperature (NDT) of the material.

ABB-CE re-evaluated the alternate method proposed in the CESSAR-DC for establishing the fracture toughness of the actual flywheel material. In Amendment N to the CESSAR-DC, ABB-CE has modified the CESSAR-DC to specify the SRP Section 5.4.1.1, Subsection II.2 recommended requirements as noted below:

The minimum static fracture toughness of the material at the normal operating temperature of the flywheel will be equivalent to a critical stress intensity factor, K_{IC} , of at least $165 \text{ MPa}\sqrt{\text{m}}$ ($150 \text{ ksi}\sqrt{\text{in.}}$). Compliance will be demonstrated by either of the following:

- testing of the actual material of the flywheel to establish the K_{IC} value at the normal operating temperature
- determining that the normal operating temperature is at least 56°C (100°F) above the RT_{NDT}

These requirements should offer adequate assurance that the material chosen will have the specified service life. On this basis, DSER Open Item 5.4.1.1-2 is resolved.

In SRP Section 5.4.1.1, the staff recommends that the normal operating temperature of the flywheel be at least 56°C (100°F) above the RT_{NDT} . The flywheel's normal operating temperature depends on the specific motor design; however, the typical flywheel normal operating temperature is 49°C (120°F). This places the temperature at least 60°C (110°F) above the specified minimum RT_{NDT} of -12°C (10°F). Additionally, in Amendment N to the CESSAR-DC, ABB-CE revised the CESSAR-DC to specify the recommended requirements in SRP Section 5.4.1.1, Subsection II.2 as noted in resolution of Open Item 5.4.1.1-2 above.

These requirements should offer adequate assurance that the material chosen will have the specified service life. On this basis, DSER Open Item 5.4.1.1-3 is resolved.

ABB-CE stated that the flywheel is subjected to a magnetic particle or liquid penetrant examination per ASME Code Section III before final assembly. Further, ABB-CE stated that the inspection be performed on finished machined bores, keyways, and on both flat surfaces to a radial distance of 20 cm (8 in.) minimum beyond the final largest machine bore diameter, but not including small drilled holes.

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In Amendment N to the CESSAR-DC, ABB-CE modified the CESSAR-DC to require that a surface examination be performed on all finished machined bores, keyways, splines, and drilled holes. This will offer assurance that these critical areas will be sound and defect free. On this basis, DSER Open Item 5.4.1.1-4 is resolved.

Originally, ABB-CE had not submitted the basis for the assumed design overspeed for staff review as recommended in SRP Section 5.4.1.1. ABB-CE described the design speed as 125 percent of normal operating speed. In Amendment N to the CESSAR-DC, ABB-CE modified the CESSAR-DC to state that the design overspeed will be at least 10 percent above the highest anticipated overspeed of the pump. The highest anticipated overspeed is predicted on the basis of the largest break size remaining after application of the leak-before-break principle of the piping as described in Section 3.6 of this report. In Amendment T to the CESSAR-DC, ABB-CE identified the largest pipe remaining after leak-before-break as a 4-inch pressure spray line. On this basis, DSER Open Item 5.4.1.1-5 is resolved.

In Amendment N to the CESSAR-DC, ABB-CE modified the CESSAR-DC to require a surface examination of all exposed surfaces as part of the flywheel ISI program. On this basis, DSER Open Item 5.4.1.1-6 is resolved.

In Amendment N to the CESSAR-DC, ABB-CE modified the CESSAR-DC and committed to performing a preservice baseline inspection incorporating all the procedures for ISI as described to establish initial flywheel conditions, accessibility, and practicality of the program. ABB-CE also specified that the examination procedures and acceptance criteria are those contained in ASME Code Section III. On this basis, DSER Open Item 5.4.1.1-7 is resolved.

In Amendment N to the CESSAR-DC, ABB-CE modified the CESSAR-DC to commit to conform to the guidance of RG 1.14, "Reactor Coolant Pump Flywheel Integrity." On this basis, DSER Open Item 5.4.1.1-8 is resolved.

The staff concludes that the precautions taken to ensure the integrity of the reactor coolant pump flywheels are acceptable and comply with the requirements of GDC 1 and 4, and 10 CFR 50.55a(a)(1). This conclusion is based on the following:

- ABB-CE's selection of materials, fracture toughness tests, design procedures, preservice overspeed spin testing program, and ISI program for the reactor

coolant pump flywheels comply with the requirements for GDC 1 and 10 CFR 50.55a(a)(1) with respect to providing adequate assurance of a quality product commensurate with the importance of the safety function.

- ABB-CE has complied with the requirements of GDC 4 and the guidance of RG 1.14, "Reactor Coolant Pump Flywheel Integrity," in using suitable materials with adequate fracture toughness and conservative design procedures, and by providing a preservice testing and ISI program for flywheels of reactor coolant pump motors which provides reasonable assurance of the structural integrity of the flywheels in the event of design overspeed transients or postulated accidents.

5.4.2 Steam Generator Materials

The staff reviewed the SG materials in accordance with SRP Section 5.4.2.1. Among the areas of review were selection and fabrication of materials, SG design, compatibility of the SG components with the primary and secondary coolant, and cleanup of the secondary side.

Reactor coolant enters at the bottom of the SG through the single inlet nozzle, flows through the U-tubes, and leaves through the two outlet nozzles. The SG tubes are fabricated of thermally treated Inconel 690 materials. The outer diameter of a tube is 1.9 cm (3/4 in.) and the wall thickness is 0.11 cm (0.042 in.). The tube support structure is of the "egg crate" design and is fabricated from stainless steel. SG components will be constructed in accordance with the ASME Boiler & Pressure Vessel Code, Section III.

In CESSAR-DC Table 5.2-2, ABB-CE lists the materials selected for the fabrication of the SG. In CESSAR-DC Table 5.2-1, ABB-CE lists the applicable ASME Code requirements. The primary side of the SG is designed and fabricated to comply with ASME Section III Class 1 criteria. The secondary side pressure boundary parts of the SG are designed and fabricated to comply with ASME Section III Class 2 criteria. The staff evaluated fabrication of the RCPB in Section 5.2.3 of this chapter.

ABB-CE stated that the SG internal components are designed to maintain localized fluid velocities below the critical velocities that will cause excessive tube vibration. The horizontal egg crate tube supports are designed and located to maintain the natural frequency of the tubes higher than the exciting frequency induced by cross-flow in the tube bundle entrance region. In comparison to the System 80 SGs, several design improvements, described below, have been made to the System 80+ SGs to pre-

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vent the type of vibration-induced wear seen at other plants:

- The economizer divider plate and center stay cylinder have been extended up to the top of the cold side recirculating fluid entrance window, to reduce fluid cross-flow velocities.
- The downcomer partition between hot side and cold side has been extended down to the top of the economizer divider plate to prevent circumferential flow.
- The economizer flow shroud has been extended up 41 cm (16 in.) to meet the downcomer flow shroud, for a circumferential distance of 18 cm (7 in.) on each side of the cold side recirculating fluid entrance window, to limit the cross-flow velocities in the open tube lane along the economizer divider plate.
- One additional egg crate tube support has been added to the cold leg economizer region to stiffen the tubes. The combination of stiffer tubes and lower local fluid velocities results in a reduction in flow-induced tube vibration for the System 80+ SGs.

A flow-induced vibration test has been performed on a test model for the SG economizer and lower tube bundle region. This test model incorporated the design improvements described above for the System 80+ SGs. In the CESSAR-DC, ABB-CE states that the results of the economizer test should provide comparable results for the System 80+ units and, therefore, excessive tube vibration due to flow-induced vibrations should not occur.

In the CESSAR-DC, ABB-CE states that the System 80+ design includes several provisions for detecting and monitoring primary-to-secondary leakage:

- condenser steam air ejector radiation monitor
- SG blowdown sample line radiation monitors
- main steam line area radiation monitor

Detection of increasing primary-to-secondary leakage is enhanced by the Nuplex 80+ control complex. Nuplex 80+ includes the capability to perform trending analysis on the radiation monitor output signals.

In the CESSAR-DC, ABB-CE asserts that the tube expansion into the tubesheet is total with no voids or crevices along the length of the tube in the tubesheet. The tubing is expanded in the tubesheet using an explosive charge. Tolerances on charge length and strength are

specified so that the crevice between tube and tubesheet is closed through the full depth of the tubesheet and so that no crevice occurs on the secondary face of the tubesheet. Therefore, the crevice between the tubesheet and the inserted tube is minimal because the tubes are expanded to the full depth of insertion of the tube in the tubesheet. The tube expansion and subsequent positive contact pressure between the tube and the tubesheet preclude a buildup of impurities from forming in the crevice region and reduce the probability of crevice boiling.

To prevent the buildup of potential corrosive materials in the tube-to-tube support plate area, extensive crevices and conditions which promote dryout should be avoided. In the CESSAR-DC, ABB-CE claims that with only one exception, all tube supports present a flat surface to the tube in order to prevent dryout at the support locations. The one exception is located in an area in which the secondary water is subcooled and, therefore, dryout is not expected. The egg crate support structures form an open lattice which can provide almost complete washing of the tube surface with SG water and thus minimizes the potential for dryout. In summary, the tube support structure is manufactured to the egg crate design. The egg crate design eliminates the narrow annular gap at the tube supports, because the support may contact the tube at only four lines on the tube circumference, and provides almost complete washing of the tube surface with SG water.

ABB-CE has incorporated the following design features to provide access to the SG internals so that tools may be inserted to inspect and clean up deposits:

- One 53-cm (21-in.) manway on each of the hot and cold leg side of the primary side. Manway locations were chosen with respect to use of remote manipulators for inspection and maintenance.
- Two 53-cm (21-in.) manways on the secondary side to allow access to the separator dryer area, and an internal hatch for access to the top of the tube bundle. These openings allow for inspections on the condition of separation equipment, feedwater ring, and the top of the tube bundle.
- Two 20-cm (8-in.) handholes at the top of the tubesheet for sludge lancing as well as for inspection of the downcomer annulus. These handholes can be used to remotely inspect for, and retrieve, loose parts.

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In response to Q281-37, ABB-CE stated that the secondary water chemistry complies with EPRI Report NP-6239, "PWR Secondary Water Chemistry Guidelines," December 1988, Revision 2. In order to comply with the secondary water chemistry guidelines, ABB-CE has incorporated the following design features and controls:

- monitoring the feedwater to limit the amount of impurities that can be introduced into the SG
- continuous blowdown of the SG to reduce the concentrating effects of the SG
- chemical addition points for establishing and maintaining an environment that minimizes system corrosion
- preoperational cleaning of the feedwater system
- minimizing feedwater oxygen content before entry into the SG

Furthermore, ABB-CE has proposed the following design and operational features in order to control feed train and SG corrosion: (1) alkaline conditions are maintained in the feed train and SG to decrease the release of soluble corrosion products from metal surfaces, thereby, resulting in the formation of a protective metal oxide film; (2) hydrazine, added to scavenge oxygen within the steam cycle, also promotes formation of a metal oxide film by reducing ferric oxide to magnetite that provides further corrosion resistance; (3) oxygen-induced corrosion is minimized by removing the oxygen from the steam cycle condensate in the main condenser deaerating section and by the full-flow feedwater deaerator which is a portion of the low-pressure feedwater heaters; and (4) the amount of oxygen within the steam cycle is also minimized by adding hydrazine which scavenges the oxygen that does enter the system. ABB-CE additionally states that oxygen-induced corrosion is minimized by providing specific materials for components such as the steam reheater tubes, feedwater heater tubes, condenser tubes, condenser tubesheets, and other miscellaneous piping.

The staff's open items in the DSER were resolved as follows:

In a letter of November 24, 1992, ABB-CE discussed how it will perform weld qualifications, weld fabrication processes, and inspection during fabrication and assembly of the SGs. This is acceptable to the staff as the procedures and processes would be in accordance with

an applicable ASME code which had been approved by the staff. On this basis, DSER Open Item 5.4.2-1 is resolved.

The COL applicant will be required to establish an ISI program for SG tubes by System 80+ TS SR 3.4.12.2. This program will be subject to review in the normal review processing of any COL application, and no further action need be taken until that time. On this basis, DSER Open Item 5.4.2-2 is resolved.

ABB-CE indicated that the secondary water chemistry will comply with EPRI guidelines. However, the staff has found several inconsistencies between the CESSAR-DC and the EPRI secondary water chemistry guidelines. ABB-CE clarified some of these inconsistencies in various submittals, and revised the CESSAR-DC with Amendments N and O as follows:

- ABB-CE stated that the normal hydrazine operating specification will not be revised to be greater than three times the oxygen level because ABB-CE believes 20 ppb is the minimum acceptable hydrazine level in order to accommodate minor fluctuations in condensate dissolved oxygen level. Whereas a lower limit of 20 ppb of hydrazine to scavenge oxygen would be adequate for low oxygen levels, a typical value of greater than three times the oxygen level would be appropriate for cases in which there are high levels of oxygen in the feedwater. In Amendment O to the CESSAR-DC, ABB-CE revised the CESSAR-DC to require that the normal concentration of feedwater hydrazine shall be greater than, or equal to, three times the dissolved oxygen concentration ($\geq 3 \times [\text{O}_2]$), but no less than 20 ppb. On this basis, DSER Open Item 5.4.2-3/1 is resolved.
- ABB-CE has changed the temperature at which chemistry guidelines should be implemented for cold shutdown/wet layup and heatup from 93 °C (200 °F) to 99 °C (210 °F). This change was made for consistency with the definition of plant operating modes specified in the TS in Chapter 16 of the CESSAR-DC. On this basis, DSER Open Item 5.4.2-3/2 is resolved.
- ABB-CE referenced cyclohexylamine as one possible additive to maintain secondary water pH. Cyclohexylamine is not referenced in EPRI guidelines as a pH additive. In Amendment O to the CESSAR-DC, ABB-CE revised Section 10.3.5.1 to state that only ammonia and morpholine will be used for pH control. On this basis, DSER Open Item 5.4.2-3/3 is resolved.

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- Minimum sampling frequencies were not specified for any of the water chemistry control parameters. In Amendment N to the CESSAR-DC, ABB-CE revised the CESSAR-DC to specify minimum sampling frequencies of secondary water chemistry in Tables 10.3.5-4 and 10.3.5-5. On this basis, DSER Open Item 5.4.2-3/4 is resolved.
- ABB-CE has limited the cold, wet, layup hydrazine concentration to 200 ppm. ABB-CE submitted the following technical basis for this limit. Concentration of hydrazine during cold, wet, layup is limited to 200 ppm because by its thermal decomposition, ammonia is produced. High hydrazine concentrations would result in excessive production of ammonia and elevated pH conditions (> 9.2) during plant heatup. Such conditions can accelerate copper alloy corrosion in secondary systems, and rapidly exhaust condensate polisher resin. The staff finds that this is an acceptable technical basis for the cold, wet, layup hydrazine concentration limit. On this basis, DSER Open Item 5.4.2-3/5 is resolved.
- The requirement for immediate shutdown if the oxygen concentration in the condensate system exceeds 30 ppb for more than 100 hours had been deleted without any technical justification being given. ABB-CE submitted as a technical justification that the CESSAR-DC follows the recommendations of the EPRI "PWR Secondary Water Chemistry Guidelines, Revision 2, December 1988" which contains no immediate shutdown limit (Action Level 3) for condensate dissolved oxygen. The staff has accepted these guidelines. On this basis, DSER Open Item 5.4.2-3/6 is resolved.
- No specification for hydrazine was presented for the makeup water used to fill the SGs when outage durations are expected to be less than seven days from cold shutdown to startup. In Amendment O to the CESSAR-DC, ABB-CE revised the CESSAR-DC to specify that for outages of less than seven days from cold shutdown to startup, the SGs will be filled with the feedwater containing more than 5 ppm hydrazine, and nitrogen overpressure greater than 5 psig. On this basis, DSER Open Item 5.4.2-3/7 is resolved.
- No requirement was specified for monitoring hydrazine feed rate when sampling for hydrazine is not possible. In Amendment N to the CESSAR-DC, ABB-CE revised CESSAR-DC Table 10.3.5-5 to provide for monitoring hydrazine feed rate when sampling for hydrazine is not possible. On this basis, DSER Open Item 5.4.2-3/8 is resolved.

- In the CESSAR-DC, ABB-CE has proposed verifying hydrazine concentration downstream of the normal chemical addition point rather than at the condensate pump discharge. The CESSAR-DC follows the recommendations of the EPRI, "PWR Secondary Water Chemistry Guidelines," Revision 2, December 1988, which has been accepted by the staff. Hydrazine concentration will be verified at a point downstream of the normal chemical addition point rather than at pump discharge because, due to the high volatility of hydrazine, the concentration would not be representative at the pump discharge. This explanation is acceptable to the staff. On this basis, DSER Open Item 5.4.2-3/9 is resolved.

In response to DSER Open Item 5.4.2-4, ABB-CE provided clarifications for the following statements relating to the secondary water chemistry:

- In the CESSAR-DC, ABB-CE states that the normal chemistry conditions can be maintained by any plant operating with little or no condenser leakage. ABB-CE clarified the meaning of this statement as follows: With All Volatile Treatment (AVT), even small leakage of condenser cooling water could significantly change secondary water chemistry, because AVT control has no buffering capability to minimize the impact of ionic impurities. The staff finds this statement acceptable.
- In CESSAR-DC Section 10.3, ABB-CE states that the normal pH value for a mixed system would be typically within 8.5 and 9.2, with action required only if operating experience shows increased copper transport at a pH greater than 9.2. In the CESSAR-DC, ABB-CE defines increased copper transport as any statistically significant increase above the base value established at pH 9.2 or below. Increased copper transport could be determined through weekly measurements of the feedwater copper concentration during normal power operation in the secondary systems containing copper alloys. The staff finds ABB-CE has provided an acceptable definition and means of controlling increased copper transport.

On the basis of these two clarifications, DSER Open Item 5.4.2-4 is resolved.

The specifications for makeup water to the feedwater and condensate system are presented in Table 9.2.3-1 of the CESSAR-DC. In Amendments J and L, ABB-CE modified this table so that all water chemistry limits complied with the EPRI evolutionary plant requirements. On this basis, DSER Open Item 5.4.2-5 is resolved.

In the CESSAR-DC, ABB-CE restricts the use of SA 540 Grade B23 or B24 bolts in the RCPB to Class 3 material which has a maximum ultimate tensile strength of 1,191 MPa (170 ksi). The use of this class of material conforms to the requirements of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." Laboratory and field experience with these materials has indicated resistance to stress corrosion cracking under anticipated service conditions. On this basis, DSER Open Item 5.4.2-6 is resolved.

ABB-CE has provided a corrosion allowance for a 60-year plant design life and its technical basis as follows:

The 60-year minimum corrosion allowance for the materials to exposed process fluids are:

- 0.00015 cm (0.0006 in.) for Inconel 690
- 0.0076 cm (0.003 in.) for austenitic stainless steel
- 0.015 cm (0.006 in.) for ferritic stainless steel
- 0.15 cm (0.060 in.) for unclad carbon steel

The materials of construction for the primary-side components include a variety of nickel alloys, stainless steels, and low-alloy steels clad with corrosion-resistant material (i.e., nickel alloy or stainless steel). All of these materials are resistant to general corrosion in high-temperature aqueous environments, and numerous studies have documented this resistance (EPRI, NP-3274, 1984, Volume 5, Page 7-44; and EPRI, NP-3274, 1984, Volume 3, page 7-42). For example, 300-series stainless steels, low-chromium stainless steels (Types 405 and 409), and Alloy 600 (a nickel-base alloy used for SG tubes and penetrations in older PWRs) have been exposed to low-level faulted secondary-side environments for up to 249 days. After exposure, the following general corrosion rates were determined: For Type 347 stainless steel, 0.000033 - 0.00012 cm/yr (0.013 - 0.048 mils/yr); for Types 405 and 409 steel, 0.00007 - 0.00015 cm/yr (0.028 - 0.061 mils/yr); and for Alloy 600, 0.000015 - 0.000058 cm/yr (0.006 - 0.023 mils/yr). Another report (A. J. Sedriks, J. W. Schultz, and M. A. Cordovi, "Inconel 690, A New Corrosion Resistant Material," Boshoku Gijutsu, Volume 28, page 82-95, 1979) presents general corrosion results for Alloy 690 and Alloy 600 in flowing (5.5 m/sec) ammoniated and borated water at 316 °C (600 °F). This study shows a significantly improved general corrosion resistance for Alloy 690 as compared to Alloy 600. Under the test conditions, general corrosion for Alloy 690 was one fourth to one tenth that of Alloy 600. Thus, a conservative corrosion allowance for Alloy 690 is 0.01 mils/yr of operation. For austenitic stainless steel in the

primary circuit, a corrosion allowance of 0.00013 cm/yr (0.05 mils/yr) is appropriate; and for ferritic stainless steels, a corrosion allowance of 0.00025 cm/year (0.1 mils/yr) is appropriate.

On the secondary side, unclad carbon and low-alloy steels used for the SG shells are exposed to the secondary environment, as are the outside surfaces of the tubes (Alloy 690), the tube supports (Type 409 stainless steel), and the flow distribution plate (Type 405 stainless steel). The same corrosion allowances used for the primary-side applications are appropriate for the secondary side for Alloy 690 and the Types 405/409 stainless steels. EPRI NP-3271 and NP-3274, along with other sources, show relatively low corrosion rates for carbon and low-alloy steels. For 1010 carbon steel, corrosion rates after up to 440 days' exposure to low-level faulted conditions ranged from 0.00034 to 0.002 cm/yr (0.135 - 0.77 mils/yr). For SA 508 Class 3, the corrosion rates for the same test period varied from 0.00031 to 0.0017 cm/yr (0.122 to 0.650 mils/yr). Thus, for secondary-side applications, a corrosion allowance of 0.0025 cm/yr (1.0 mils/yr) of operation is reasonable, and an overall corrosion allowance of 0.15 cm (0.060 in.) will provide sufficient material for 60 years of operation. On this basis, DSER Open Item 5.4.2-7 is resolved.

The concern relating to the environmental effects on fatigue was discussed in Section 5.2.3 of this chapter. This is applicable to all materials.

The design of System 80+ components will address the potential influence of environmental effects on the fatigue life of materials over the 60-year design life. The issue of environmental effects on fatigue is currently under consideration by a special Steering Committee for Cyclic Life and Environmental Effects in Nuclear Applications of the PVRC. These activities were initiated in response to requests from the ASME B&PV Code Committee and the BNCS. The charter of the PVRC Steering Committee is to provide guidance and direction related to determining the effects of light-water-reactor (LWR) service environments on the cyclic life properties of applicable materials. The steering committee is also evaluating application methodologies that include these effects in the fatigue analysis process.

Preliminary recommendations were made to the BNCS in September 1992. The initial findings reported to BNCS were that the current S/N curves should be appropriate for PWR environments. There was not a complete agreement of the steering committee on this position and the issue is unresolved. ABB-CE will continue to monitor the industry activities on the fatigue curves and fatigue analysis methodology.

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System 80+ components will be designed to the ASME B&PV Code requirements. ABB-CE addressed the issue of environmental fatigue of Class 1 components in Amendment R to Section 3.9.1 of CESSAR-DC and the staff's evaluation is provided in Section 3.12.5.7 of this report. On this basis, DSER Open Item 5.2.3-6 is resolved.

In Section 5.2.3 of this chapter, the staff notes that the ferrite content limits for austenitic stainless steel given in the CESSAR-DC are broader than those in industry guidelines and staff guidance. In Amendment Q to the CESSAR-DC, ABB-CE modified the CESSAR-DC to state that ferrite content of austenitic stainless steel castings with normal operating temperatures above 500 °F (260 °C) will be limited to a maximum of 20 percent. In Amendment L to the CESSAR-DC, ABB-CE modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes the lower ferrite content limits for austenitic stainless steel castings and weld metal will provide reasonable assurance that components of these materials maintain adequate fracture toughness for their 60 year life. On this basis, Open Item 5.4.2-9 is resolved.

As discussed in Sections 4.5.1, 4.5.2, and 5.2.3 of this report, ABB-CE limits the carbon content of austenitic stainless steels to 0.065 percent compared to the usual specification limit of 0.08 percent and imposed controls on welding, fabrication, and water chemistry during startup and operation to mitigate the occurrence of IGSCC in these stainless steels. These mitigating actions have proved successful in operating reactors for more than 15 years and offer reasonable assurance that IGSCC can be controlled, and therefore, maintain the integrity of these components. On this basis, DSER Open Item 5.4.2-10 is resolved.

Operating experience indicates that SGs may have to be replaced during plant life. In a letter of December 18, 1992, ABB-CE described plant design provisions to facilitate SG replacement. The descriptions and procedures for removing and replacing the SGs demonstrated an adequate approach for their removal. On this basis, DSER Open Item 5.4.2-11 is resolved.

The staff concludes that the SG materials specified are acceptable and meet the requirements of GDC 1, 14, 15, and 31, and Appendix B to 10 CFR Part 50. This conclusion is based on the following:

- ABB-CE has complied with the requirements of GDC 1 with respect to codes and standards by ensuring that the materials selected for use in Class 1 and Class 2 components will be fabricated and inspected

in conformance with codes, standards, and specifications acceptable to the staff. Welding qualification, fabrication, and inspection during manufacture and assembly of the SGs will be done in conformance with the requirements of Sections III and IX of the ASME Code.

- The requirements of GDC 14 and 15 have been met to ensure that the reactor coolant boundary and associated auxiliary systems will be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture during normal operation and anticipated operational occurrences.

The primary side of the SG is designed and fabricated to comply with ASME Code Section III, Class 1 criteria as required by the staff. The secondary-side pressure boundary parts of the SG are designed and fabricated to comply with ASME Code Section III, Class 2 criteria.

- The crevice between the tubesheet and the inserted tube is minimal because the tube expands to the full depth of insertion of the tube in the tubesheet. The tube expansion and subsequent positive contact pressure between the tube and the tubesheet preclude a buildup of impurities in the crevice region and reduce the probability of crevice boiling.

The tube support structure is manufactured to the egg crate design. The egg crate design eliminates the narrow annular gap at the tube supports, because the support may contact the tube at only four lines on the tube circumference, and almost completely washes the tube surface with SG water.

- The requirements of GDC 31 have been met with respect to the fracture toughness of the ferritic materials, since the pressure boundary materials of ASME Code Class 1 components of the SG comply with the fracture toughness requirements and tests of Subarticle NB-2300 of Section III.
- The requirements of Appendix B to 10 CFR Part 50 have been met since the onsite cleaning and cleanliness controls during fabrication will conform to the recommendations of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The controls placed on the secondary coolant chemistry are in agreement with staff technical positions.

- Reasonable assurance of the satisfactory performance of SG tubing and other generator materials is offered by (1) the design provisions and the manufacturing requirements of the ASME Code, (2) rigorous secondary water monitoring and control, and (3) the limiting of condenser in-leakage. The controls described above, combined with conformance to applicable codes, standards, staff positions, and RGs, constitute an acceptable basis for meeting the requirements of GDC 1, 14, 15, and 31, and Appendix B to 10 CFR Part 50 as it relates to chemical and materials engineering aspects.

5.4.3 Shutdown Cooling System

In CESSAR-DC Section 5.4.7, ABB-CE describes the shutdown cooling system (SCS). The SCS is designed to remove heat from the RCS during a reactor shutdown after the RCS temperature and pressure have been reduced to approximately 177 °C (350 °F) and 3,150 kPa (450 psia). The SCS is capable of reducing the RCS temperature to the refueling conditions and maintaining this temperature until the plant is started again.

The SCS also performs the following functions:

- transfers RCS fluid to the CVCS for purification of RCS fluid during SCS operation
- transfers refueling pool water back to the inside-containment IRWST following refueling operations
- provides cooling of the IRWST during post-accident feed and bleed operations utilizing the SIS and the safety depressurization system
- provides RCS LTOP

The staff reviewed the SCS for the System 80+ design in accordance with SRP Section 5.4.7. The design acceptance criteria are stated in BTP RSB 5-1, "Design Requirements of the Residual Heat Removal System." ABB-CE's approach to meeting these requirements is discussed below.

5.4.3.1 Functional Requirements

In SRP Section 5.4.7, the staff requires the SCS design to meet GDCs 1 through 5. Compliance with GDC 1 through 4 is addressed in Chapter 3 of this report. ABB-CE has complied with GDC 5, since components of the SCS will not be shared between units.

When the SCS is in operation, the system takes its suction from each hot leg through a system of parallel lines

and valves forming redundant trains. From the discharge of the two SCS pumps, a portion of the reactor coolant is circulated through two SCS heat exchangers, which are cooled by component cooling water. The reactor coolant then returns to the RCS through SIS DVI nozzles. During normal shutdown, when non-safety-related equipment and offsite power are available, decay heat is removed from the core by the main feedwater system, the steam bypass system, and reactor coolant pump circulation system. During emergency shutdown, when non-safety-related equipment and offsite power are not available, decay heat is removed by RCS natural circulation with the SGs serving as the heat sink. The steam produced in the SG shell side can be removed through the safety-related SG safety valves and atmospheric steam dump valves to vent vaporized secondary coolant. Secondary coolant makeup is the emergency feedwater pumped from two safety-grade emergency feedwater storage tanks.

During natural circulation conditions, RCS depressurization can be achieved by one of three systems: (1) auxiliary pressurizer spray, (2) rapid depressurization system (RDS), and (3) reactor coolant gas vent system (RCGVS). The auxiliary pressurizer spray is not designed to satisfy single-failure criteria and is not credited in the BTP RSB 5-1 analysis. Both RDS and RCGVS are safety-grade systems. The RCGVS is designed for RCS depressurization for design-basis events, and the RDS is used to mitigate consequences of a beyond-design-basis event. The staff reviewed the design of the RCGVS and RDS in Section 6.7 of this report.

Two separate trains for each unit provide redundancy in the SCS. Each train is powered by an emergency diesel generator. No single active failure to the SCS can prevent at least one complete train of the SCS from being brought on line from the control room during normal plant cooldown, a transient, or an accident.

Before SCS initiation, the SI tanks must be secured or vented (from the control room) to prevent overpressurization of the SCS. During the cooldown, adding borated water (boration) to the RCS controls core reactivity. For cold shutdown, boron is added to the RCS using the SIS. The source of borated water is the fluid in the IRWST. As opposed to the System 80 design, the System 80+ design does not rely on the CVCS as a means for boron injection to meet BTP RSB 5-1 requirements.

Item G of BTP RSB 5-1 requires that a seismic Category I emergency feedwater supply be provided with sufficient inventory to permit operation at hot shutdown

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conditions for at least 4 hours, followed by a cooldown to the conditions permitting operation of the SCS. The emergency feedwater needed for the cooldown is based on the longest cooldown time needed with either onsite or offsite power available and with the worst single failure. ABB-CE includes two safety-grade emergency feedwater trains in the System 80+ design. Each train has one turbine-driven pump and motor-driven pump. The emergency feedwater system contains two safety-grade emergency feedwater storage tanks. Each tank has a minimum condensate volume of 1.3×10^6 L (350,000 gal) available for safe plant cooldown.

During the course of the review, the staff asked ABB-CE to submit the results of an analysis to demonstrate that the System 80+ plant can achieve cold shutdown during natural circulation conditions per the assumptions specified in BTP RSB 5-1. In the response to RAI Q440.51, ABB-CE referenced the CE System 80 report of August 12, 1983 (letter LD-83-074; Docket No.: STN 50-470F), as applicable to the System 80+ design. The CE report of August 12, 1983, contains an analysis of full natural circulation cooldown (NCC) from hot standby conditions to temperatures and pressures that permit the initiation of the SCS. The analysis was performed using only safety-related equipment concurrent with a loss-of-offsite power and the worst single failure (i.e., failure of one diesel generator to start). The results of this analysis indicate that the total time required to take the System 80 plants from hot standby conditions to the SCS initiation condition is approximately 10.5 hours. This time includes maintaining the plant in hot standby for 4 hours before commencing a cooldown.

In the report of August 12, 1983, for the System 80 analysis, the design used the safety-related auxiliary spray and charging systems for RCS depressurization and RCS inventory control. The safety-related RCGVS was not credited in the analysis. During this simulated cooldown process, approximately 832,800 L (220,000 gal) of condensate water are required, which is less than the 1.1×10^6 L (300,000 gal) design condensate water capacity. ABB-CE compared the operational status of systems and equipment assumed in the analysis and indicated that the availability of the system and components for System 80+ is equivalent to System 80, and the capacities of those components for plant cooldown are equivalent to or better than those for System 80. In the System 80+ design, the auxiliary spray and charging systems are not designed to be single-failure-proof and are not creditable for the BTP RSB 5-1 analysis. However, under the assumption of loss-of-offsite power, two trains of the SIS are available for System 80+ as opposed to one train of the high-pressure SIS for System 80. In response to the staff's RAI, ABB-CE claims that

the flow capacity of the additional SI train for System 80+ exceeds that of the charging pumps credited in the System 80 analysis for the RCS coolant makeup. In the System 80 cooldown analysis, the auxiliary spray was used to rapidly depressurize the RCS. This depressurization can be conducted for System 80+ via the RCGVS, which has greater capability for depressurization than the System 80 auxiliary spray system. In addition, the condensate storage capability has more than doubled from the System 80 to System 80+ design. Thus, ABB-CE asserts that the August 12, 1983 analysis for System 80 bounds the System 80+ design.

ABB-CE's justification was reasonable qualitatively. However, the staff was concerned about the thermal-hydraulic response during depressurization by using the RCGVS. Also, since the makeup water is demanded at relatively high RCS pressures during the plant cooldown, the SI pumps may not be as effective as the charging pumps for the RCS inventory makeup. The staff determined that ABB-CE had not adequately demonstrated the applicability of the August 12, 1983, analysis to the System 80+ design. Therefore, ABB-CE was required to submit the results of analysis demonstrating that System 80+ is capable of achieving cold shutdown in accordance with the assumptions specified in BTP RSB 5-1.

In response to the staff's request, ABB-CE performed an NCC analysis by using only safety-grade equipment with an assumption of the concurrent loss of offsite power and the worst single-failure. The results are presented in Amendment N to Appendix 5D of the CESSAR-DC. ABB-CE used the LTC code that NRC had approved earlier to perform the analysis. The NCC sequence relied on the guidance in ABB-CE's System 80+ emergency operating guidelines (EOGs) as follows:

- Following the reactor trip, the operator manually controls the atmospheric dump valves (ADVs) and emergency feedwater pumps to restore the plant at hot standby conditions.
- The operator throttles two of the four SIS pumps (only two trains are available because of the assumed single failure of one diesel generator) for RCS boration and inventory control.
- After the 4-hour hot standby period, the operator uses the ADVs to initiate a cooldown with a rate less than that allowed by the TS.
- The operator uses the safety-grade equipment for RCS pressure and inventory control: the pressurizer vent system for depressurization; the reactor vessel vent system for removing steam from the reactor

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upper head; and SI pump throttling for boron and RCS inventory control.

- The operator controls the cooldown to maintain the pressurizer level and RCS subcooling within the ranges consistent with the EOG guidance.

The analytical results indicate that the SCS entry conditions can be achieved within 12 hours (including 4 hours at hot standby conditions). The staff has reviewed ABB-CE's NCC analysis. The staff finds that the previously approved LTC code was used for analysis and only the safety-grade equipment was credited for NCC. The analysis shows that the SCS entry conditions have been achieved with the total emergency feedwater usage of less than 35 percent of the minimum available capacity. Therefore, the staff concludes that the analysis is acceptable. BTP RSB 5-1 requires ABB-CE to demonstrate the NCC capability by analysis and test. The staff believes that the use of RCGVS and SIS may result in a more complicated activity than that demonstrated by use of auxiliary spray and CVCS. The staff's conclusion is based upon the SI pump curve which shows a rapidly increasing flow rate as pressure decreases, and the use of two vent valves for pressure control as opposed to the auxiliary spray system. To satisfactorily resolve this open item, the staff required ABB-CE to conduct an NCC test to confirm that the operator in the control room can control the plant cooldown by using the methods consistent with the methods assumed in the analysis.

In response, ABB-CE addressed the operating margins relative to the NCC analysis, identified the instrumentation available to the operator in the control room for carrying out the NCC operation, discussed tests for validation of the NCC analysis, and compared the RCS depressurization rates achieved using the pressurizer vent versus the auxiliary spray. Pressurizer venting will be used in the System 80+ design, although auxiliary spray was used in the NCC test for the Palo Verde plant. The ABB-CE evaluation was included in Appendix 5D to CESSAR-DC Amendment U.

In CESSAR-DC Appendix 5D, ABB-CE indicated that the NCC analysis presented in meeting BTP RSB 5-1 used 35 percent of the minimum condensate available. This excess condensate capacity provides a large operating margin. It allows additional time for the operator to extend the cooldown duration. For example, the plant cooldown rate could be reduced from the value used in the NCC analysis, resulting in a milder thermal transient on plant components. The additional time could also be used to control the SI flow injected into the reactor vessel by throttling the SI valves. This action would

result in a milder pressure transient while maintaining control of the reactor coolant makeup function.

The operator is required to monitor the following parameters during the NCC event: (1) pressurizer level, (2) pressurizer pressure, (3) hot-leg temperature, (4) cold-leg temperature, (5) SG pressure, (6) SG level, (7) reactor vessel upper head level, and (8) emergency feedwater tank level. In CESSAR-DC Appendix 5D, ABB-CE indicates that the instrumentation used by the operator to measure these parameters is safety grade and consists of multiple channels as defined in Table 3.3.14-1 of CESSAR-DC Chapter 16, thus complying with BTP RSB 5-1 requirements which only allow the use of safety-grade equipment.

ABB-CE also agreed to perform tests, which were discussed in Table 5D-3 of Appendix 5D and Chapter 14 of the CESSAR-DC, to validate the adequacy of the assumptions used in the NCC analysis in meeting BTP RSB 5-1 requirements. The tests include: (1) pressurizer vent and reactor vessel vent capacity tests, (2) SI flow capacity and valve throttling capability tests, (3) atmospheric dump valve and emergency feedwater capacity tests, and (4) tests of the decay power to reactor coolant flow ratio and boron mixing in NCC conditions using the auxiliary spray for depressurization. ABB-CE compared the analytical results of the RCS depressurization rate using the pressurizer vent, which is safety-grade equipment, versus auxiliary spray (which was used to achieve the NCC for the Palo Verde plant for meeting the BTP RSB 5-1 requirements.) ABB-CE indicated that the depressurization rates for the auxiliary spray and pressurizer vent are similar with a slightly higher depressurization rate achieved when using the auxiliary spray. The system responses are similar. After completing the analytical results, ABB-CE asserted that both methods will provide adequate depressurization capacities for the NCC event.

The staff has reviewed Appendix 5D of the CESSAR-DC and finds that (1) ABB-CE demonstrated that the NCC analysis is conservative, and sufficient operating margins are available for the operator to achieve the NCC cooldown, (2) the multiple channels and safety-grade instrumentation will be available to the operator for monitoring and controlling the NCC conditions from the control room, (3) the tests proposed by ABB-CE represent anticipated plant conditions during the NCC event, and the acceptance criteria of the tests bound the assumptions used in the NCC analysis, and (4) the pressurizer vent capacity for depressurization will be demonstrated to be comparable to that of the auxiliary spray, which was used in a Palo Verde plant with a RCS simi-

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lar to the System 80+ design for the NCC test in meeting the BTP RSB 5-1 requirements. Therefore, the staff concludes that ABB-CE has provided reasonable assurance that the NCC can be achieved for System 80+ plants and ABB-CE's approach to addressing the NCC issue is acceptable. On this basis, DSER Open Item 5.4.3.1-1 is resolved.

Since System 80+ differs from System 80 in the DVI feature, ABB-CE committed to conduct a boron mixing test under natural circulation for the first plant with a System 80+ design. In a letter of December 23, 1992, ABB-CE indicated that the test would be performed during the startup test program as described in Amendment J to CESSAR-DC Section 14.2.12.4.23. Since the boron mixing test would be performed after fuel was loaded during power ascension testing, the staff agrees with ABB-CE that this test should not be part of the ITAAC program, which would be completed before fuel load. On this basis, DSER Open Item 5.4.3.1-2 is resolved.

The SCS is a safety-grade system, is single failure proof, and all of the actions necessary for shutdown can be performed from the control room. Therefore, the System 80+ SCS complies with the functional requirements of BTP RSB 5-1.

5.4.3.2 SCS Isolation

The SCS is designed to provide adequate protection to the SCS from overpressurization when the RCS is operating at high pressures. There are two parallel paths of SCS suction lines that are equipped with three remotely controlled valves in each line. Two isolation valves per line are located inside the containment and one motor-operated valve per line is located in the low-pressure piping outside the containment. The four isolation valves inside the containment are powered by four independent power supplies, so that a fault in a power supply or valve will neither line up the RCS to either of the two SCS trains inadvertently nor prevent the initiation of shutdown cooling with at least one SCS train. Each of the four isolation valves inside the containment has an independent interlock, utilizing pressurizer pressure, to prevent the SCS isolation valves from being opened when reactor pressure is high. This interlock protects against violation of the low temperature overpressure protection (LTOP) and plant pressure and temperature limits (see Section 5.2.2 of this chapter for the staff evaluation of the LTOP design). The interlocks do not provide automatic closure of the valves when pressurizer pressure increases during shutdown cooling operation, so

as not to increase the potential for loss of shutdown cooling. The discharge side is isolated from the RCS by three check valves on each line from the SCS discharge to the DVI nozzles.

ABB-CE confirmed that valve position indication for each of four isolation valves inside the containment and two motor-operated valves outside the containment will be installed in the main control room and that CESSAR-DC Figure 6.3.2.1C correctly reflects the valve position and indications for each isolation valve. On this basis, DSER Open Item 5.4.3.2-1 is resolved.

The design of interlocks will not automatically close the isolation valves in the event of an RCS pressurization during shutdown cooling. This design of interlocks prevents a loss of decay heat removal capability due to inadvertent closure of the isolation valves during SCS operation. To respond to DSER Open Item 5.4.3.2-2, ABB-CE indicated that the System 80+ design has the following features for preventing SCS overpressurization:

- An alarm is installed for each of the SCS suction isolation valves located inside containment. This alarm warns control room operators when rising RCS pressure approaches the SCS operating pressure limit and isolation valves are not closed.
- Valve position indication is installed in the control room for all SCS suction isolation valves located inside containment. The power for the indicators is supplied from a separate source so that the position indication is not affected by power interruption to the operator.
- The response guidelines as described in CESSAR-DC Section 5.4.7.2.6.E direct operator actions during shutdown cooling for overpressurization protection of the SCS.
- The design pressure of the SCS is increased to 6,205 kPa (900 psi). This design pressure meets the required ultimate rupture strength (URS) equal to full RCS pressure as specified in SECY-90-016, "Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements," to reduce risk of intersystem loss-of-coolant accident (LOCA) events (see Chapter 20, of this report (Issue 105), for the evaluation of intersystems LOCA events).

On the basis of its review, the staff finds that the design features discussed above provide reasonable assurance of overpressure protection for the SCS. The staff, thus, determines that the design of interlocks without auto-

matic closure function is acceptable. On this basis, DSER Open Item 5.4.3.2-2 is resolved.

ABB-CE indicated that the SCS design deviates from the requirements of BTP RSB 5-1, Items B.1(b) and (c), and Item E with regard to the diversity of interlocks for SCS suction isolation valves, and testability of the isolation valves and interlock circuits. In Item B.1(b) of BTP RSB 5-1, the staff states that "the valves shall have independent diverse interlocks to prevent the valves from being opened unless the RCS pressure is below residual heat removal (RHR) system design pressure." ABB-CE justified the lack of this diversity by addressing the adequacy of the overall design of the SCS: overpressure protection described in CESSAR-DC Section 5.4.7.2.3; the LTOP addressed in CESSAR-DC Section 5.2.2.10; the alarm design for operator action detailed in CESSAR-DC Section 5.4.7.2, and the limiting conditions for operation and SRs for the relief valves stated in TS 3.4.11. The staff has reviewed these areas for the SCS design. On the basis of its evaluation stated in Sections 5.2.2 and 5.4.3 of this chapter, the staff concludes that reasonable assurance was provided to protect against SCS overpressurization, even without diverse interlocks.

In Item B.1(c) of BTP RSB 5-1, the staff states that "the valves shall have independent diverse interlocks to protect against one or both valves being open during an RCS increase above the design pressure of the RHR system." To address the overpressure concern in this item, previous designs have contained an interlock that automatically closes the suction isolation valves upon a pressure transient for PWRs. Over the past several years, there has been increased effort to improve the reliability of the SCS. It was recognized that the auto closure interlocks (ACIs) on suction valves of the SCS have been a frequent cause of loss-of-SCS events. The staff had previously reviewed and approved the removal of the ACIs for many PWRs. The System 80+ design does not have the ACI feature, which is consistent with the modified design in the existing plants, and the guidance specified in EPRI URD 5.2.3.3. As discussed in this section, the staff has concluded that the overall design of the SCS system is adequate for overpressure protection. Therefore, the staff finds that the deviation from the guidance of Item B.1(c) is acceptable.

In Item E of BTP RSB 5-1, the staff states that "the isolation valve operability and interlock circuits must be designed so as to permit on line test when operating in the RHR modes." ABB-CE has retracted this deviation and committed to the conformance of the System 80+ design to these testing requirements. Testing of the SCS

suction isolation valve circuitry appears in CESSAR-DC Sections 7.6.2.2 and 14.2.12.1.21.

The SCS design complies with the isolation requirements of BTP RSB 5-1 and is, therefore, acceptable. On this basis, DSER Open Item 5.4.3.2-3 is resolved.

Intersystems LOCA

SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," specifies the staff's position on protection against the possibility of a LOCA occurring outside the containment for those systems linked to the RCS. The staff position is that future advanced light-water reactor (ALWR) designs should reduce the possibility of a LOCA outside the containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an URS at least equal to full RCS pressure. The "extent practicable" phrase is a realization that all systems must eventually interface with atmospheric pressure, and that, for certain large tanks and heat exchangers, it would be difficult or prohibitively expensive to design such systems to the URS equal to full RCS pressure. Note that the degree of isolation or number of barriers (e.g., three isolation valves) is not sufficient justification for using low-pressure components that can be practically designed to the URS criteria. For example, piping runs should always be designed to meet the URS criteria, as should all associated flanges, connectors, and packing, including valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drains and vent lines. The designer should attempt to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

For all interconnected systems and components that do not fully comply with all RCS URS criteria, it is necessary for ABB-CE to justify why it is not practical to reduce the pressure challenge any further. This justification should be based on an engineering feasibility analysis and not solely on risk-benefit tradeoffs.

For those interconnections where acceptable justification on impracticability of full RCS pressure capability has been provided, compensating isolation capability must be demonstrated. For example, ABB-CE should demonstrate for each connecting system and component that the degree and quality of isolation or reduced severity of the potential pressure challenges compensate for and justify the safety of the low-pressure system or component.

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The adequacy of pressure relief and piping of relief back to primary containment are possible considerations. As noted in SECY-90-016, each of these areas where high pressure meets low pressure must also include the following protection measures: (1) the capability of leak testing the pressure isolation valves, (2) the valve position indication that is available in the control room when isolation valve operators are deenergized, and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low-pressure system and both isolation valves are not closed.

In Amendment Q to the CESSAR-DC, ABB-CE submitted its evaluation of various interconnected systems (including the SCS design) in Appendix 5E, and concluded that the design of the SCS that is face-to-face with the RCS will be in full compliance with the requirements specified in SECY-90-016 as discussed for interfacing system LOCA protection. The staff has reviewed the SCS design in meeting the intersystems LOCA requirements. On the basis of its evaluation in Chapter 20 of this report on GSI-105, the staff finds that ABB-CE adequately addressed the intersystems LOCA requirements for the SCS design, and concludes that the SCS design is acceptable. On this basis, DSER Open Item 5.4.3.2-4 is resolved.

5.4.3.3 SCS Pressure Relief

Each of two SCS suction lines is equipped with pressure relief valves designed to prevent overpressurization of the SCS. Relief valves SI-169 and SI-469 are in the SCS suction line between the isolation valves inside the containment to protect isolated pipe lengths from transient thermal effects. Relief valves SI-179 and SI-189 are in suction lines between isolation valves inside the containment and isolation valves outside the containment. These two valves protect the SCS from inadvertent RCS pressurization during SCS operation. The valves are sized to protect the components and piping from overpressurization due to inadvertently starting the charging pumps, RCS pumps, SI pumps, and pressurizer heaters. The evaluation of the design for the SCS relief valves is discussed in Section 5.2.2 of this chapter.

The SCS piping and valves from the RCS up to and including SI-653 and SI-654, are designed to ASME Code Section III, Class 1. The remainder of the SCS piping, components, and valves, including relief valves SI-179 and SI-189, are designed to ASME Code Section III, Class 2. The relief valves SI-179 and SI-189 are located inside the containment. The reactor coolant discharged through the relief valves is collected in the holdup volume tank (HVT), which is a low collection

point in the containment. The spillage from the HVT is collected in the IRWST, which supplies the borated water for the SIS. This relief flow path enables the System 80+ design to preserve the RCS inventory and the SIS water source in the containment, and to avoid flooding of any safety-related equipment should relief valves be stuck open.

The SCS design complies with the pressure relief requirements of BTP RSB 5-1 and is, therefore, acceptable.

5.4.3.4 SCS Pump Protection

Each of the SCS pumps, as described in CESSAR-DC Section 5.4.7.2.2.E and in ABB-CE's response to RAI Q440.58, has a minimum recirculation line to protect the pump from a potential low-flow or no-flow operating condition. The miniflow lines are routed from the pump discharge back to the pump suction.

A locally operated manual valve, that is located in each miniflow line to allow pump maintenance, is locked open during all operating modes. A heat exchanger in each miniflow line removes pump heat in the event of a closed pump discharge path should an operator make a mistake.

Individual flow and pump inlet/outlet pressure instruments monitor the condition of each of the SCS pump trains. A readout for each of these instruments is in the main control room. In addition, the SCS flow has a low-flow alarm located in the main control room. An alarm alerts the operator to low-flow conditions that lead to a loss of shutdown cooling due to either a loss of adequate pump suction or the closure of a system valve.

In a letter dated January 21, 1993, and in Amendment V to CESSAR-DC Sections 5.4.7.2.2.E and 5.4.7.4, ABB-CE indicated that the SCS pumps are required to operate in a range from the design point to runout conditions. These pumps are not required to operate at reduced flow, such as injection mode following a small-break LOCA. For post-trip long-term cooling, the SCS operates only after the RCS level has been stabilized, and the pressure and temperature have been reduced to low pressures. Since SCS pumps take suction from RCS hot legs and return flow to the RCS through a DVI nozzle, RCS pressure has no effect on SCS pump flow. ABB-CE reviewed operating data for pumps with the similar mechanical seal design used in SCS pumps and stated that these pumps could be operated at the range of the design flow to runout conditions for more than 12,000 hours without overhaul. ABB-CE also indicated that the

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SCS pumps will be inspected, tested, repaired, and replaced to comply with the pump functional requirements in accordance with the requirements of ASME/ANSI OM-1987. On the basis of its evaluation of the SCS design discussed in this section, the staff concludes that ABB-CE has given reasonable assurance for the operability of SCS pumps. On this basis, DSER Open Item 5.4.3.4-1 is resolved.

The SCS design conforms to the pump protection requirements of BTP RSB 5-1; therefore, the SCS pump design is acceptable.

5.4.3.5 Shutdown and Low-Power Operation Risk

The NRC staff has become increasingly concerned about the safety of operations during low-power operations or periods of plant shutdown. The Diablo Canyon event of April 10, 1987, highlighted a particularly sensitive condition of the operation of a PWR with a reduced inventory in the RCS. After the NRC reviewed the event, the staff issued Generic Letter (GL) 88-17, "Loss of Decay Heat Removal," on October 17, 1988. The letter requested that licensees address numerous generic deficiencies to improve operational safety at reduced reactor coolant inventory. This included deficiencies in procedures, hardware, and training in the areas of (1) prevention of accident initiation, (2) early mitigation of accidents, and (3) control of radioactive material if a core damage accident should occur. More recently, the loss of ac power at the Vogtle plant on March 20, 1990, led to a staff investigation of the incident. The incident investigation team report (NUREG-1410), "Loss of Vital AC Power and the Residual Heat Removal System During Midloop Operations at Vogtle Unit 1 on March 20, 1990," June 1990, emphasized the need for risk management of shutdown operations.

These events have led the staff to conclude that: (1) non-routine activities and availability of less equipment during shutdown increases the probability of complex events that challenge operators in unfamiliar ways and (2) lack of rigorous consideration of accident sequences during shutdown operations has resulted in potentially incomplete or inadequate instrumentation, emergency response procedures, and mitigation equipment.

The NRC has evaluated risks during shutdown and low power operation. This task had the objective of developing a thorough understanding of the manner in which activities and operations during shutdown are planned and implemented; determining the root causes of earlier events was also a part of this task. The staff evaluation of shutdown and low-power operations was issued in

NUREG-1449, "NRC Staff Evaluation of Shutdown and Low Power Operation," September 1993.

Because an event that occurs during shutdown and at low power has safety significance, the staff has determined that concerns with the shutdown operation should be satisfactorily addressed before the NRC issues a final design approval for the System 80+ design. ABB-CE was required to demonstrate adequate treatment of shutdown risk by submitting information, including (1) documentation showing adequate assessment of shutdown and low-power risk, with identification of design-specific vulnerabilities and weaknesses and (2) documentation showing consideration and incorporation of design features that minimize shutdown and low-power risk vulnerabilities. The staff has reviewed the shutdown risk report submitted by ABB-CE and found it acceptable. The staff's evaluation is included Section 19.3 of this report. On this basis, DSER Open Item 5.4.3.5-1 is resolved.

In addition, the staff asked ABB-CE to describe any deviations and a justification of the deviations from GL 88-17 recommendations regarding instrumentation, operating procedures, operator training programs, and TS implementations, addressing the concerns for reduced inventory operation. The staff sent its request to ABB-CE in RAIs Q440.140 through Q440.151 on August 21, 1991. The staff has reviewed ABB-CE's responses to the RAIs in conjunction with the shutdown-risk report. The staff finds that ABB-CE adequately addressed the shutdown-risk issue. The staff's evaluation of the shutdown risk appears in Section 19.3 of this report. On this basis, DSER Open Item 5.4.3.5-2 is resolved.

Tests, Operational Procedures, and Support Systems

In the CESSAR-DC, ABB-CE requires that preoperational tests be conducted on each referenced plant to verify proper operation of the SCS. The preoperational tests include calibration of instrumentation, verification of adequate cooling flow, and verification of operability of all associated valves. In addition, a preoperational hot functional performance test is performed on the installed shutdown cooling heat exchangers to verify adequate flow. In compliance with RG 1.68, "Initial Test Programs for Water Cooled Reactor Power Plants," plans to conduct preoperational testing to verify the design performances of the SCS and its individual components. Also, preoperational hydrostatic tests are performed per Section III of the ASME B&PV Code. The preoperational testing program is evaluated in Chapter 14 of this report.

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Conclusions

The staff reviewed the SCS design for the System 80+ design. The scope of the staff review included piping and instrumentation diagrams, failure modes and effect analysis, and design performance specifications for essential components. The staff reviewed the proposed design criteria, design bases, and the conformance of design to those criteria and bases. The staff concludes that the SCS design for the System 80+ plant complies with the requirements of BTP RSB 5-1; GDC 5, "Sharing of Structures, Systems, and Components"; GDC 19, "Control Room"; and GDC 34, "Residual Heat Removal"; and is acceptable. The design description and ITAAC for the SCS system contain the appropriate design commitments that are to be verified and that are acceptable.

5.4.4 Pressurizer Relief Tank

The IRWST is used as the pressurizer relief tank for the System 80+ design and is addressed in Section 6.8 of this report.

6 ENGINEERED SAFETY FEATURES

6.1 Materials

6.1.1 Engineered Safety Feature Materials

The staff has reviewed the engineered safety features (ESF) materials, fabrication procedures and associated requirements stated in the CESSAR-DC for compliance with the guidelines in standard review plan (SRP) Section 6.1.1. Among the areas reviewed were the materials and fabrication procedures used in the design ESFs, composition and compatibility of ESFs fluids, component and system cleaning, thermal insulation, and coatings. Open items in the draft safety evaluation report (DSER) are addressed as follows:

In CESSAR-DC Section 5.2.3.3.2.1, ABB-CE takes exception to the recommendations in Position C.2 of Regulatory Guide (RG) 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," for controls imposed on preheat temperatures for welding ferritic steels. These controls provide reasonable assurance that components made from low-alloy steels will not crack during fabrication and will minimize the possibility of subsequent cracking because of hydrogen-induced stresses in the weldment. In a submittal dated November 24, 1992, ABB-CE stated that its basis for taking exception to Position C.2 was the Westinghouse Topical Report WCAP-8678, "Effect of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels," September 1975. The NRC evaluated this report and accepted it. This report presents three acceptable alternatives, and ABB-CE's position is that a particular alternative will be specified when purchased in the construction stage on the basis of such factors as the configuration or the capabilities of the fabrication facility. This approach should provide adequate assurance that low-alloy steel weldments will not develop cracking from the presence of hydrogen. On this basis, DSER Open Item 6.1-1 is resolved.

In CESSAR-DC Table 6.1-1, ABB-CE indicates that Inconel 690 will be used in place of Inconel 600 as an ESF pressure-retaining material. Operating experience indicates that Inconel 600 is susceptible to cracking as discussed at a 1987 workshop on secondary-side intergranular corrosion mechanisms and documented in EPRI NP-5971. ABB-CE has considered using alternate materials that resist stress corrosion cracking. ABB-CE will use Inconel 690 wherever Inconel 600 was to be used. The staff views the Inconel 690 alloy as the preferred nickel alloy in the primary and secondary coolant loops because it is more resistant to corrosion than Inconel 600. The use of Inconel 690 will provide reasonable assurance of the material integrity of the components and tubing in contact with reactor coolant and secondary water chemistries thus

meeting the EPRI guidelines committed to by ABB-CE. On this basis, DSER Open Item 6.1-2 is resolved.

ABB-CE proposes that Types 304 and 316 austenitic stainless steel be used. However, these materials are susceptible to intergranular stress corrosion cracking when the oxygen content of the reactor coolant exceeds 0.010 ppm at temperatures above 93 °C (200 °F) during normal operations. During startup and operation of the nuclear steam supply system (NSSS), these temperature and chemical conditions are maintained through specified chemistry control. ABB-CE has taken alternative mitigating approaches described in CESSAR-DC Section 5.2.3.4 as allowed in RG 1.44, "Control of the Use of Sensitized Stainless Steel." These approaches provide reasonable assurance of the integrity of austenitic stainless steel components in contact with reactor coolant. On this basis, DSER Open Item 6.1-3 is resolved.

The ferrite content limits for austenitic stainless steel castings and weld metal in the System 80+ design are broader than those in industry guidelines Electric Power Research Institute (EPRI), "Advanced Light Water Reactor Utility Requirements Document," NP-6780-L, Volume 2, Advanced Light Water Reactor (ALWR) Evolutionary Plant, Chapter 1, Overall Requirements, Revision 3, November 1991, and staff guidance U.S. Nuclear Regulatory Commission, "Technical Report on Material Selection and Processing Guidelines for (BWR) Coolant Pressure Boundary Piping," NUREG-0313, Rev. 2, January 1988. In Amendment Q to CESSAR-DC, ABB-CE committed to limit the ferrite content of austenitic stainless steel castings to a maximum of 20 percent for components with normal operating temperature exposure above 260 °C (500 °F). In Amendment L to CESSAR-DC, ABB-CE modified the upper ferrite limit to 15 percent for austenitic stainless steel weld metal. The staff believes that these lower limits at ferrite content for austenitic stainless steel castings and weld metal will provide reasonable assurance that components of these materials will maintain adequate fracture toughness for their 60-year life. On this basis, DSER Open Item 6.1-4 is resolved.

During the development of the DSER, the staff raised the issue that carbon steel materials may be susceptible to dynamic strain aging (DSA) which reduces the material fracture properties (see C.W. Marschall, M.P. Landow, and G.M. Wilkowski, "Effect of Dynamic Strain Aging on Fracture Resistance of Carbon Steels Operating at Light-Water-Reactor Temperatures," ASTM STP 1074, ASTM, Philadelphia, PA, 1990, pp. 339-360.) The NRC staff discussed this issue with ABB-CE several times. In its January 20, 1993, submittal, ABB-CE noted that industry is investigating the susceptibility of materials to DSA and the extent to which the fracture toughness properties are

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affected. The materials used in nuclear power plant ESFs (carbon and alloy steels, stainless steels and nickel alloys) have adequate fracture toughness, either as an inherent property of the material (austenitic stainless steels, thin-section carbon and alloy steels, and nickel alloys) or by Charpy V-notch testing of thicker section carbon and alloy steels as required by the American Society of Mechanical Engineers (ASME) Code. These materials have had extensive use in existing reactor designs in the same applications and have had no cases of excessive leakage, sudden failure or brittle fracture due to DSA. The staff concludes that these materials, by meeting the ASME Code, have an acceptable level of fracture toughness to account for DSA. On this basis, DSER Open Item 6.1-5 is resolved.

In Section 5.2.3 of the CESSAR-DC, ABB-CE is proposing to allow the continued reference to American Society for Testing and Materials (ASTM) A-708 for the purpose of maintaining the qualifications of older welding procedures. ASTM A-262, Practice A or E (recommended in RG 1.44) has been used by ABB-CE since the mid-1970s for verifying non-sensitization of austenitic stainless steel materials and weldments. ABB-CE qualified weld procedures developed prior to the mid-1970s and used A-708 in their qualification. These weld procedures are still in use. There have been no intergranular stress corrosion cracking (IGSCC) failures involving austenitic stainless steel weldments in ABB-CE's NSSS units. In addition, in SRP 4.5.1, paragraph III.2, the staff allowed A-708 as an acceptable alternative test for A-262, Practice A or E. On the basis, DSER Open Item 6.1-6 is resolved.

ABB-CE limits the carbon content of austenitic stainless steels to 0.065 percent compared to the usual specification limit of 0.08 percent and imposed controls on welding, fabrication, and water chemistry during startup and operation to mitigate the occurrence of IGSCC in these stainless steels. Such mitigation has proved successful in operating reactors for more than 20 years and provides reasonable assurance that IGSCC is controlled and, therefore, that these components will maintain their integrity. On this basis, DSER Open Item 6.1-7 is resolved.

ABB-CE has provided a corrosion allowance for a 60-year plant design life and its technical basis as follows:

The 60-year minimum corrosion allowance for the materials exposed to process fluids are:

0.00015 cm (0.0006 in)	for Inconel 690
0.0076 cm (0.003 in)	for austenitic stainless steel
0.015 cm (0.006 in)	for ferritic stainless steel

0.15 cm (0.060 in) for unclad carbon steel

Among the materials of construction for the primary-side components are a variety of nickel alloys, stainless steels, and low alloy steels clad with corrosion resistant material (i.e., nickel alloy or stainless steel). All of these materials are resistant to general corrosion in high temperature aqueous environments, and numerous studies have documented this resistance (EPRI URD and NUREG-0313). For example, 300 series stainless steels, low-chromium stainless steels (Types 405 and 409), and Alloy 600 (a nickel alloy used for steam generator (SG) tubes and penetrations in older pressurized-water reactors (PWRs) have been exposed to low-level faulted secondary-side environments for up to 249 days. After exposure, the following general corrosion rates were determined: for Type 347 stainless steel, 0.000033 – 0.00012 cm/year (0.013 – 0.048 mils/yr); for Type 405/409 stainless steel, 0.00007 – 0.00015 cm/year (0.028 – 0.061 mils/yr); and for Alloy 600, 0.000015 – 0.000058 cm/year (0.006 – 0.023 mils/yr). ASTM STP 1074 presents general corrosion results for Alloy 690 and Alloy 600 in flowing (5.5 m/sec) ammoniated and borated water at 316 °C (600 °F). This study shows a significantly improved general corrosion resistance for Alloy 690 over Alloy 600. Under the test conditions, general corrosion for Alloy 690 was one fourth to one tenth that of Alloy 600. Thus, a conservative corrosion allowance for Alloy 690 is 0.000025 cm/year (0.01 mils/yr) of operation. For austenitic stainless steel in the primary circuit, a corrosion allowance of 0.00013 cm/year (0.05 mils/yr) is appropriate; and for ferritic stainless steels, a corrosion allowance of 0.00025 cm/year (0.1 mils/yr) is appropriate.

On the secondary side, unclad carbon and low-alloy steels used for the steam generator shells are exposed to the secondary environment, as are the outside surfaces of the tubes (Alloy 690), the tube supports (Type 409 stainless steel), and the flow distribution plate (Type 405 stainless steel). The same corrosion allowances used for the primary-side applications are appropriate for the secondary side for Alloy 690 and the Types 405/409 stainless steels. References (1) and (2), along with other sources, show relatively low corrosion rates for carbon and low-alloy steels. For Type 1010 carbon steel, corrosion rates after up to 440 days of exposure to low-level faulted conditions ranged from 0.00034 – 0.002 cm/year (0.135 – 0.77 mils/yr). For SA 508 Class 3 steel, the corrosion rates for the same test period varied from 0.00031 to 0.0017 cm/year (0.122 – 0.650 mils/yr). Thus, for secondary-side applications, a corrosion allowance of 0.0025 cm/year (1.0 mils/year) of operation is reasonable, and an overall corrosion allowance of 0.15 cm (0.060 in.) will provide sufficient material for 60 years of operation. On this basis, DSER Open Item 6.1-8 is resolved.

Recent fatigue test data indicates that the effects of the environment could significantly reduce the fatigue resistance of materials (see the following references). (1) K. Iida, J. Fukakura, M. Higuchi, H. Kobayashi, S. Miyazono, and M. Nakao, "Survey of Fatigue Strength Data of Nuclear Structural Materials in Japan," Abstract of DBA Committee Report, 1988. (Presented to the ASME, Subgroup on Fatigue Strength, on December 5, 1988, in New York City, NY.) (Enclosure in letter, from John W. Craig of NRC to Edward Griffing of the Nuclear Management and Resources Council, Washington, D.C., dated July 2, 1991.). (2) M. Higuchi and K. Iida, "Fatigue Strength Correction Factors of Carbon and Low-Alloy Steels in Oxygen-Containing High-Temperature Water," Nuclear Engineering and Design, Volume 129, 1992, pp. 293-306. (3) J.B. Terrell, "Effect of Cyclic Frequency on the Fatigue Life of ASME SA-106-C Piping Steel in PWR Environments," Journal of Materials Engineering, Volume 10, Number 3, 1988, pp. 193-203). The specific concerns relate to the water and temperature environment and its synergistic interactions with the strain rate.

The design of System 80+ components will address the potential influence of environmental effects on the fatigue life of materials over the 60-year design life. The issue of environmental effects on fatigue is currently under consideration by a special Steering Committee for Cyclic Life and Environmental Effects in Nuclear Applications of the Pressure Vessel Research Council (PVRC). These activities were initiated based on requests from the ASME Boiler and Pressure Vessel (B&PV) Code Committee and the Board on Nuclear Codes and Standards (BNCS). The charter of the PVRC Steering Committee is to provide guidance and directions related to determining the effects of light water reactor (LWR) service environments on the cyclic life properties of applicable materials. The Steering Committee is also evaluating application methodologies that include these effects in the fatigue analysis process.

Preliminary recommendations were provided to the BNCS in September 1992. The initial findings reported to BNCS were that the current S/N curves should be appropriate for PWR environments. There was not a complete agreement of the Steering Committee on this position and the issue is not yet finally resolved. ABB-CE will continue to monitor the industry activities on the fatigue curves and fatigue analysis methodology.

System 80+ components will be designed to ASME B&PV Code rules. If the influence of environmental effects has not been incorporated into the Code rules at the time of the design, the potential effects will be addressed based on the technical understanding of the materials data and anticipated operating conditions. On this basis, DSER Open Item 6.1.9 is resolved.

The staff concludes that the ESF materials specified are acceptable and meet the requirements of General Design Criteria (GDC) 1, 4, 14, 31, 35, and 41 of Appendix A to 10 CFR Part 50; Appendix B to 10 CFR Part 50, and 10 CFR 50.55a. The basis for this conclusion follows.

The facility will comply with GDC 1, 14 and 31, and 10 CFR 50.55a regarding assuring an extremely low probability of leakage, of rapidly propagating failure, and of gross rupture. This is shown since the materials selected for ESFs will conform to Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the Code. Cold-worked stainless steels will not be used for components of the ESF. The fracture toughness of the ferritic materials will conform to the requirements of the code.

The controls on the use and fabrication of the austenitic stainless steel of the systems will follow the recommendations of RG 1.31, "Control of Ferrite Content of Stainless Steel Weld Metal," and RG 1.44, "Control of the Use of Sensitized Stainless Steel," or acceptable alternative approaches. Fabrication and heat treatment practices performed in accordance with these guidelines produce added assurance that the probability of IGSCC will be reduced during the postulated duration of the accident.

Conformance with the codes and RGs and with the staff positions given above, constitutes an acceptable basis for fulfilling the requirements of GDC 1, 4, 14, 35, and 41, Appendix B to 10 CFR Part 50, and 10 CFR 50.55a, and for designing, fabricating, and erecting these systems so that they can function as required.

GDC 1, 14, and 31 and Appendix B to 10 CFR Part 50 have been met with respect to assuring that the reactor coolant boundary and associated auxiliary systems will have an extremely low probability of leakage, rapidly propagating failures, and gross rupture. The controls to be placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the ESFs conform to recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Compliance with the recommendations of RG 1.36 form a basis for conforming to the requirements of GDC 1, 14, and 31.

The protective coating systems have been qualified by tests acceptable to the staff. This qualification provides reasonable assurance that the coating systems will not degrade the operation of the ESFs by delaminating, flaking, or peeling.

The coatings applied are in accordance with RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." Confor-

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mance with this RG is a basis for meeting the requirements of Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants."

The ESF components will meet the requirements of GDC 4, 35, and 41, and Appendix B to 10 CFR Part 50 regarding compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). The controls on the pH and chemistry of the reactor containment sprays and the emergency core cooling water following the loss-of-coolant or design-basis accident (DBA) are adequate to reduce the probability of IGSCC of austenitic stainless steel components and welds of the ESFs systems in containment throughout the duration of the postulated accident to completion of cleanup.

Also, the control of pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, conforms to RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provides assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal or cause serious deterioration of the materials in the containment.

The staff reviewed the CESSAR-DC for compliance with the staff positions in Branch Technical Position (BTP) MTEB 6-1, "pH for Emergency Coolant Water for PWRs." The controls on the pH and chemistry of the reactor containment sprays and emergency core cooling system (ECCS) solutions conform to the staff positions on postaccident chemistry requirements for PWR emergency coolant water. They also conform to the requirements of GDC 14 for assuring the low probability of abnormal leakage or failure of the reactor coolant pressure boundary and safety-related structures. Therefore, the staff concludes that the proposed pH for emergency cooling water is acceptable.

The controls to be placed upon component and system cleanup will comply with the recommendations of RG 1.37. These controls provide a basis for the finding that the components and systems will be protected against damage or deterioration by contaminants as stated in the cleaning requirements of Appendix B to 10 CFR Part 50.

In CESSAR-DC, ABB-CE states that the combined license (COL) applicant will review vendor fabrication procedures to ensure that unstabilized austenitic stainless steel with a carbon content greater than 0.03 percent is not exposed to the temperature range of 430 °C to 820 °C (800 °F to

1500 °F) other than during welding. This was designated as COL Action Item 6.1.1-1 in the DSER.

The COL applicant will perform grinding with resin-bonded or rubber-bonded aluminum oxide or silicon carbide wheels that have been used only on Type 300-series stainless alloys. Grinding wheels bonded with rubber compositions that include halides or sulfur will not be used on austenitic stainless steels. This was designated as COL Action Item 6.1.1-2 in the DSER.

To prevent halide-induced intergranular corrosion which could occur in an aqueous environment with significant quantities of dissolved oxygen, the COL applicant will add hydrazine to inhibit the flushing water. This was designated as COL Action Item 6.1.1-3 in the DSER.

In welding ferritic steel, one of the options of Westinghouse report WCAP-8578, "Effect of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels," September 1975, will be followed as an alternative to Position C.2 of RG 1.5, "Control of Preheat Temperature for Welding of Low-Alloy Steel." Additional requirements of the ASME Code will also be followed. This was designated as COL Action Item 6.1.1-4 in the DSER.

6.1.2 Organic Materials

The staff concludes that the protective coating systems and their applications are acceptable and comply with the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on ABB-CE having complied with the quality assurance requirements of Appendix B to 10 CFR Part 50, since the coating systems and their applications will comply with the positions in RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." The quality assurance standards of American National Standards Institute (ANSI) N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities" have been superseded. The requirements of ANSI N101.2 are met by selecting coatings based on ASTM D-3842, "Standard Guide for Selection of The Methods for Coatings for Use in Light Water Nuclear Power Plants," ASTM D-3911 "Standard Test Method for Evaluating Coatings Used in Light Water Nuclear Power Plants at Simulated Design Basis Accident Conditions," and ASTM D-3843 "Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities. These ASTM specifications represent the current industry technology and experience and are intended to replace the ANSI standards referenced in ANSI N101.2. The containment coating systems have been evaluated regarding their suitability to withstand a postulated DBA environment.

The coating systems chosen by ABB-CE have been qualified under conditions that take into account the postulated DBA conditions.

6.2 Containment Systems

The containment systems for the System 80+ design comprise: (1) a containment structure as the primary containment; (2) a secondary containment (shield building) surrounding the primary containment; and (3) supporting systems. The primary containment is designed to prevent the uncontrolled release of radioactivity to the environment with a design leakage rate of 0.5 percent of free volume for the first day and 0.25 percent of free volume each day after the first 24 hours. These values are predicated on a containment pressure caused by the DBA equal to the containment's design value of 365.4 kPa (53 psig) for the first 24 hours and equal to 50 percent of its design value (i.e., 182.7 kPa, or 26.5 psig) after 24 hours. The secondary containment is discussed in Section 6.2.3 of this report.

6.2.1 Primary Containment Functional Design

The System 80+ primary containment design consists of a 61m diameter (200-ft) spherical steel shell with a nominal wall thickness of 4.45 cm (1.75 in.). This wall is thicker around primary containment penetrations to structurally compensate for these openings. The primary containment will enclose the NSSS (i.e., reactor vessel, steam generators, reactor coolant pumps, pressurizer, and associated connecting piping), the in-containment refueling water storage tank (IRWST), safety injection tanks, the refueling canal, and associated mechanical, electrical, and heating, ventilation, and air conditioning (HVAC) support components.

The primary containment shell is supported by embedding a lower segment between the concrete of the containment internal structures and the subsphere concrete of the reactor building. There is no structural connection between the freestanding portion of the containment and the adjacent structures other than penetrations and their supports. Thus, the portion of the spherical primary containment shell above the support region (elevation 91 + 9) is structurally independent.

The primary containment has a net free volume of 95,630 m³ (3,377,000 ft³) and is designed to withstand pressures and temperatures resulting from a spectrum of primary coolant and steamline pipe breaks and from a negative differential pressure caused by an inadvertent actuation of the containment spray system. The primary containment design parameters are an internal design pressure of 365.4

kPa (53.0 psig), an external design pressure of -13.8 kPa (-2.0 psig), and a design temperature of 143 °C (290 °F).

The System 80+ containment design features are compared with those of ABB-CE's other designs in Table 6.1 of this report. These features include containment structure type, power level, containment free volume, design pressures, design temperatures, and calculated peak DBA containment pressures and temperatures.

A number of insights and conclusions can be drawn from Table 6.1. The System 80+ containment design represents a significant change from some of ABB-CE's previous containment designs, which consisted of steel-lined, pre-stressed, post-tensioned concrete structures with no secondary containment.

The System 80+ free volume is considerably larger than all of ABB-CE's previous containments. Although this larger volume is expected when comparing the System 80+ design to ABB-CE's lower-power plants, the comparative ratios of containment free volume to power show that the System 80+ containment design has a considerably larger free volume to power ratio than is found in ABB-CE's other designs. This ratio is between 19.2 and 19.9 m³/MW (677 and 704 ft³/MW) for ABB-CE's earlier designs, but is 24.6 m³/MW (869 ft³/MW) for the System 80+. This larger relative volume is proportional to the relatively lower design pressure for the System 80+ (365.4 kPa, or 53 psig) as compared to ABB-CE's steel lined, pre-stressed, post-tensioned concrete containment designs (372.3 to 413.7 kPa or 54 to 60 psig).

Table 6.1 also shows that the external design pressure of the System 80+ plant (13.8 kPa or 2 psig) is less than that of the steel lined, pre-stressed, post-tensioned concrete containment System 80 and 3400 - megawatt thermal (MWt) CE designs (34.4 kPa or 5 psig). The System 80+ containment has a significantly higher design leak rate for the first 24 hours than do other designs. This design leak rate is 0.5 percent per day, while the System 80 and 3400 MWt designs have a design leak rate of 0.1 percent per day. This higher design leak rate is a benefit to a utility because it provides more operating flexibility for containment leak rate testing and associated maintenance. This is consistent with the ALWR URD requirement. The 10 CFR Part 100 limits are still met with this higher leak rate. Containment design pressure margin is discussed in Section 6.2.1.1.2 of this report.

Table 6.1 Comparison of ABB-CE containment design features

Parameter	System 80+	System 80	3400 MWt CE	2700 MWt CE
Power, MWt	3914	3800	3400	2700
Type of Containment Structure	1.75 in. thick steel sphere primary cont.; 3 ft. thick reinforced concrete cylindrical secondary cont.	0.25 in. thick steel-lined pre-stressed post-tensioned 3.5 to 4.0 ft. thick concrete cylinder	0.25 in. thick steel-lined pre-stressed post-tensioned 3.75 to 4.3 ft. thick concrete cylinder	0.25 in. thick steel-lined pre-stressed post-tensioned 3.00 to 3.75 ft. thick concrete cylinder
Secondary containment	yes	no	no	no
Free volume	(3.4 x 10 ⁶ ft ³) 9.6 x 10 ⁴ m ³	(2.6 x 10 ⁶ ft ³) 7.4 x 10 ⁴ m ³	(2.3 x 10 ⁶ ft ³) 6.5 x 10 ⁴ m ³	(1.9 x 10 ⁶ ft ³) 5.4 x 10 ⁴ m ³
Volume to power ratio (ft ³ /MW)	(869 ft ³ /MW) 24.6 m ³ /MW	(684 ft ³ /MW) 19.4 m ³ /MW	(677 ft ³ /MW) 19.2 m ³ /MW	(704 ft ³ /MW) 20.0 m ³ /MW
Internal design pressure	365.4 kPa (53 psig)	413.7 kPa (60 psig)	413.7 kPa (60 psig)	372.3 kPa (54 psig)
External design pressure	13.8 kPa (-2.0 psig)	27.6 kPa (-4.0 psig)	34.5 kPa (-5.0 psig)	n/a
Design temperature	(290 °F) 143 °C	(300 °F) 149 °C	(300 °F) 149 °C	(289 °F) 143 °C
Design leak rate, %/day; 0-24 hours > 24 hours	0.5 0.25	0.1 0.1	0.1 0.1	0.5 0.25
Calculated peak internal pressure (design margin)	331.7 kPa (48.11 psig) (~10%)	341.3 kPa (49.5 psig) (~18%)	384.0 kPa (55.7 psig) (~7%)	356.5 kPa (51.7 psig) (~4%)
Calculated peak external pressure (design margin)	-12.6 kPa (-1.82 psig) (~9%)	-17.93 kPa (-2.6 psig) (~35%)	-18.62 kPa (-2.7 psig) (~46%)	n/a

6.2.1.1 Containment Pressure and Temperature Response to High-Energy Line Breaks

The staff reviewed the temperature and pressure response of the primary containment to a spectrum of LOCAs and main steamline breaks (MSLBs), an analysis of negative or external pressure for the primary containment, and an analysis of the minimum containment backpressure for LOCA analyses. The staff addresses the response of the secondary containment in Section 6.2.3 of this report.

6.2.1.1.1 Containment Analytical Model

ABB-CE calculates the pressure and temperature response of the containment using the CONTRANS computer code. CONTRANS models both active and passive heat sinks in the containment and the energy source using information from a table of mass and energy releases. These heat sinks include containment spray and fan cooler and the materials inside the containment that can absorb energy from the containment atmosphere.

The containment is simulated as a vapor and liquid region with time incremental step thermodynamic properties solved by CONTRANS. Mass and energy can be transferred between these two regions by boiling, condensation, and evaporation. CONTRANS models noncondensable gases and allows for a different temperature in each region. Liquid condensed from the atmosphere is automatically deposited in the sump or liquid region of the model.

The staff finds all but one of the analytical models and all of the assumptions used by ABB-CE acceptable to calculate the containment pressure and temperature transients following a LOCA and an MSLB in the System 80+ containment. The staff addresses the exception concerning post-LOCA mass and energy release data in Section 6.2.1.3 of this report.

6.2.1.1.2 Containment Pressure Response

The maximum differential pressure in the primary containment occurs from an MSLB at 0 percent power with the loss of one containment spray system. The maximum differential pressure is 331.7 kPa (48.11 psig) at 262 seconds after the MSLB begins. ABB-CE determined this peak pressure after analyzing 13 different LOCA and MSLB scenarios. This value of calculated peak differential internal containment pressure provides a margin of only 1 percent to the original design pressure of 337.8 kPa (49 psig).

In Item II.a of SRP Section 6.2.1.1.A, the staff states that an applicant should maintain a 10-percent margin during the construction permit stage of review and that at the operating license (OL) stage of review, the calculated peak pressure "should be approximately the same as at the construction permit stage." In Item II.2 of SRP Section 6.2.1.1.A, the staff cites GDC 50 as requiring "sufficient margin" for calculated peak containment pressure. Therefore, the staff stated in the DSER that the one percent margin between calculated peak primary containment pressure and its design value did not comply with the requirements given in GDC 50 or SRP Section 6.2.1.1.A. This was designated as DSER Open Item 6.2.1.1.2-1.

In CESSAR-DC, ABB-CE states that the single-step licensing process for System 80+ precludes considering this review at the construction permit (CP) stage. They also pointed out that relaxing the 10 CFR Part 100 requirement for considering the Level B operating-basis earthquake (OBE) would provide additional margin in containment pressure. However, the staff stated that the System 80+ design will not be considered final until construction and operation. Experience shows us that numerous

factors such as design changes, as-built deviations, and analytical developments have affected other plants, prompting applicants to reanalyze the containment response to LOCAs and MSLBs and increase the calculated peak containment pressure. Furthermore, a "final design" should incorporate a sufficiently large margin in peak calculated containment pressure to accommodate such future changes that might increase the calculated peak containment pressure.

The staff stated in the DSER that ABB-CE should not rely on a possible future licensing action such as a change in 10 CFR Part 100. The staff cannot predict the manner in which future changes in nuclear regulations will affect containment pressure margin. Therefore, the staff concluded that the margin in calculated peak internal primary containment pressure did not comply with SRP Section 6.2.1.1.A and GDC 50. In response, ABB-CE stated that the NRC has reviewed the elimination of the OBE as related to the change in 10 CFR Part 100 and found it acceptable. As a result, in Amendment N, ABB-CE revised the containment design pressure to 365.4 kPa (53 psig), which provides more than 10-percent margin. On this basis, DSER Open Item 6.2.1.1.2-1 is resolved.

ABB-CE calculated a peak external differential pressure of -12.6 kPa (-1.83 psid) on the primary containment caused by an inadvertent actuation of the containment spray system. ABB-CE assumed conservative containment initial conditions to minimize the calculated containment pressure in this calculation. This value provides a margin of 8.5 percent to the containment design value of -13.8 kPa (-2.0 psid). This margin is less than the 10-percent criterion previously discussed. This was designated as DSER Open Item 6.2.1.1.2-2. In response, ABB-CE refined the calculated peak external pressure to -12 kPa (-1.82 psid) which provided a 9.8-percent margin. As discussed above, this 10-percent margin criterion is required for the preliminary design, not for the final design. Because the difference between the calculated margin and the 10-percent criterion is insignificantly small, the staff agrees that the existing margin is adequate and, on this basis, DSER Open Item 6.2.1.1.2-2 is resolved.

In Item II.b of SRP Section 6.2.1.1.A, the staff states that to satisfy GDC 38, the containment pressure should be reduced to less than 50 percent of its peak value within 24 hours of the occurrence of a design-basis LOCA. ABB-CE presented the results of an analysis of long-term containment analysis pressures resulting from the design-basis LOCA which demonstrated the desired result. These analysis results are acceptable because they comply with the requirements of SRP Section 6.2.1.1.A Section (Item II.b).

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On the basis of its review of CESSAR-DC Sections 6.2.1.1, 6.2.1.3, and 6.2.1.4, the staff finds that the short-term calculated peak containment internal and external pressures comply with the requirements given in the SRP and GDC 50. The methods used in calculating these pressures are acceptable and suitably conservative, except for the one concern that is discussed in Section 6.2.1.3 of this report. The long-term pressure analysis is acceptable and conforms with the requirements in the SRP.

6.2.1.2 Subcompartment Analysis

In CESSAR-DC Section 6.2.1.2, ABB-CE states that the leak-before-break (LBB) concept applies to all piping in the primary coolant system and that, therefore, it need not analyze the subcompartments in the reactor cavity, pressurizer, and IRWST. These are the only subcompartments within the primary containment that ABB-CE lists.

The LBB concept is acceptable as a justification for not analyzing subcompartments containing primary coolant piping within the containment. The staff addresses LBB in Section 3.6.3 of this report. In the DSER, the staff stated it accepts the use of the LBB concept to justify not analyzing the subcompartments in the reactor cavity, pressurizer, and IRWST for breaks in the primary coolant piping. However, ABB-CE had not accounted for a feedwater line break in the steam generator subcompartment. The staff stated that ABB-CE may not have found all the high-energy line breaks for which LBB may not apply and which may affect subcompartment pressurization.

The System 80+ design includes a subcompartment around the steam generators through which the main feedwater line (MFWL) enters and connects to the steam generator. In the DSER, the staff stated that the MFWL is a high-energy pipe-line that could justify performing a subcompartment analysis of the steam generator enclosure. ABB-CE did not discuss this subcompartment and did not present a justification or technical evaluation to demonstrate that an MFWL would not break before leaking. ABB-CE did not indicate that it had found and reviewed all possible line breaks. Therefore, the staff required that ABB-CE must list all the high-energy lines in each subcompartment and justify not performing subcompartment analysis for each by discussing the applicability of LBB for each line. This was designated as DSER Open Item 6.2.1.2-1. ABB-CE replied that the area around the steam generator has extensive openings so that the compartment pressurization does not pose a concern. ABB-CE revised CESSAR-DC Section 6.2.1.2, which identified all the subcompartments within the containment.

In addition, ABB-CE stated that a pipe break in these compartments would increase pressure negligibly because

of adequate venting. The staff concurs with ABB-CE's assessment and, on this basis, DSER Open Item 6.2.1.2-1 is resolved.

6.2.1.3 Postulated LOCA Mass and Energy Release

ABB-CE calculated the mass and energy release data for five LOCA breaks using the CEFLASH4A and FLOOD-MOD2 computer codes for the blowdown and reflood phases, respectively. These five LOCA pipe breaks include examples in both the hot leg and the cold leg and examples for which the safety injection flow is at minimum and maximum rates.

The staff previously approved both CEFLASH4A and FLOOD-MOD2, which ABB-CE has used extensively. Moreover, the staff cited the CEFLASH4A code in SRP Section 6.2.1.3 for calculating mass and energy release data. These codes model the primary system for mass and energy release calculations by solving equations for the conservation of mass, energy, and momentum. The calculations take the following heat sources into account: primary and secondary coolant, metal within the primary and secondary coolant systems, core power transient and decay heat, water from the safety injection system (SIS), and steam generator heat transfer. However, ABB-CE states in CESSAR-DC Section 6.2.1.3.8 that it does not consider energy from the metal-water chemical reaction. ABB-CE estimated the amount of this energy to be small and to have a very small effect on the containment pressures. The staff found that not using the metal-water energy is not conservative and is inconsistent with the requirements in GDC 50. Although the effect may be small, the staff stated in the DSER that ABB-CE should account for it because the containment pressure only had a margin of about one percent (based on the original 49 psig design pressure), as discussed in SER Section 6.2.1.1.2. Including this small energy source should not cause any difficulty but will resolve the concern of nonconservatism. Therefore, the staff determined that the metal-water energy should be included as an energy source. This was designated as DSER Open Item 6.2.1.3-1. In Amendment N, ABB-CE stated that considering the increased energy from the metal-water reaction for a nominal core power of 3914 MWt, the calculated peak containment pressure would only increase to 46.7 psig (322 kPa). This is below the containment design pressure, and is also below the calculated peak pressure of the limiting MSLBs. On this basis, DSER Open Item 6.2.1.3-1 is resolved.

The nuclear steam supply system (NSSS) initial conditions are all assumed in a conservative direction so as to maximize the release of mass and energy. These conditions include an additional 2 percent of initial core power, 20 percent of decay heat for the first 1000 seconds and 10

percent thereafter, and the maximum possible coolant volumes and initial temperatures.

The CEFLASH4A model of the System 80+ NSSS includes the following conservative modeling assumptions: all-metal wall surfaces, even if they may not be an effective heat source; an infinite wall surface heat transfer coefficient; conservatively high carbon steel thermal conductivity for all metal, including the lower thermal conductivity stainless steel; a high nucleate boiling heat transfer coefficient for the steam generator tube secondary side, even after turbine trip; an unrealistically small closure time for the turbine stop valve and a closure time for the large main feedwater isolation valve to provide the maximum secondary-side energy source; and no addition of cold emergency feedwater.

ABB-CE models the reflood phases with FLOOD-MOD2 and an iteratively converged containment backpressure. ABB-CE assumes conservative values for the heat transfer coefficients, exposed area, and thermal resistance of the steam generator tubes. The model also explicitly follows the carryover fraction liquid entrainment guidance in Section II.3.c. of SRP Section 6.2.1.3.

ABB-CE models the phase after reflood in a manner similar to the reflood phase except for changing the carryout rate fraction from 0.8 to 1.0 and taking no credit for the steam condensation induced by the safety injection water. These assumptions cause a maximum rate of mass and energy release during this phase.

During the long-term cooling (LTC) or decay heat phase, ABB-CE models the NSSS as a pot of boiling water in which decay heat and stored energy supply steam to the containment, and in a manner, follow the containment depressurization rate. All decay heat and stored energy produces steam.

Therefore, the staff concludes that the methodology, assumptions, and initial conditions for the LOCA mass and energy release rate calculations comply with SRP Section 6.2.1.3 and provide for a suitably conservative analysis of the parameters for the containment pressure and temperature response analysis.

6.2.1.4 Main Steamline Break Mass and Energy Release

In analyzing the temperature and pressure response of the containment, ABB-CE used the SGN-III computer code to calculate the MSLB mass and energy release rate. ABB-CE considered a spectrum of eight different breaks that included four different power levels (102, 50, 20, and 0

percent), two different break areas [0.81 and 0.42 m² (8.72 and 4.5 ft²)] and two postulated failures (loss of one containment spray train and main steam isolation valve (MSIV) failure). ABB-CE made conservative assumptions on initial conditions such as power level, steam pressure, and fluid inventories. The assumed active failures are failure of either one containment spray train or one MSIV.

SGN-III accounts for delayed closure of isolation valves, flow from the intact steam generator prior to isolation, initial pumped feedwater flow, and the later expansion of the feedwater inventory into the steam generator. The code models the secondary system by dividing it into nodes with an appropriate balance of mass within each node. SGN-III also solves the conservation of energy and momentum equations and explicitly accounts for subcooled and saturated fluid in the system. In Item II.2 of SRP Section 6.2.1.4, the staff cites SGN-III as an acceptable computer code for calculating the release of mass and energy from an MSLB.

Therefore, the staff concludes that the methods, inputs, and assumptions in CESSAR-DC Section 6.2.1.4 for calculating the MSLB mass and energy release data for containment pressure and temperature analyses comply with SRP Section 6.2.1.4.

6.2.1.5 Minimum Containment Pressure Analysis for ECCS Performance

In Appendix K to 10 CFR Part 50, SRP Section 6.2.1.1.A (Item II.h) and SRP Section 6.2.1.5, the staff requires that ABB-CE analyze the minimum containment pressure during a LOCA to determine the containment backpressure for calculating the performance of the emergency core cooling system (ECCS) during a LOCA. The assumptions and initial conditions for this calculation are delineated in BTP CSB 6-1.

ABB-CE analyzed the minimum containment pressure using the assumptions from BTP CSB 6-1, namely, minimum containment temperature and pressure, maximum containment relative humidity and free volume, and maximum heat removal capacity of containment sprays and fan coolers. ABB-CE also assumed that the containment purge system would continue operating. ABB-CE modeled the effects of spilled ECCS water inside containment as a heat sink and also modeled the passive heat sinks such as concrete and metal present in the primary containment.

For CESSAR-DC, ABB-CE presents the results of minimum containment pressure analysis in three figures, namely: containment pressure, air temperature, and sump temperature for the first 500 seconds of the LOCA. ABB-

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CE used these results in CESSAR-DC Section 6.3.3 for the ECCS performance analysis.

The staff has reviewed ABB-CE's analysis of the containment minimum pressure for ECCS performance and finds that it conforms with Appendix K to 10 CFR Part 50, SRP Section 6.2.1.5, and BTP CSB 6-1. Therefore, the analysis is acceptable.

6.2.1.6 Conclusion

The staff has reviewed ABB-CE's containment functional design, subcompartment analysis postulated LOCA mass and energy release, main steamline break mass and energy release, and minimum containment pressure analysis for ECCS performance, and finds them consistent with the guidance in SRP Sections 6.2.1, 6.2.1.1.A, 6.2.1.2, 6.2.1.3, 6.2.1.4, and 6.2.1.5, and BTP CSB 6-1. On the basis of its review, the staff concludes that ABB-CE has met the requirements of GDC 16 and 50 in Appendix A to 10 CFR Part 50 with respect to the containment functional design; that the containment design conditions are not exceeded for as long as postulated accident conditions require; and that the containment structure and its internal components can accommodate the calculated pressure and temperature conditions from any LOCA with sufficient margin. The temperature and pressure profiles in the CESSAR-DC for the spectrum of LOCA and main steamline breaks are acceptable for use in equipment qualification.

6.2.2 Containment Heat Removal Systems

In CESSAR-DC Section 6.2.2, ABB-CE cites the containment spray system (CSS) as the only containment heat removal system qualified for post-accident operation. This system consists of two 100-percent capacity redundant and independent trains, each comprising a heat exchanger, pump, spray headers and associated piping, instrumentation, and valves. The CSS is Safety Class 2, seismic Category I, and is designed to the ASME Boiler and Pressure Vessel Code Section III.

Responding to several questions from the staff on CSS pump behavior, ABB-CE detailed the calculation of minimum pump net positive suction head (NPSH) and design features that prevent debris from entering the pumps while ensuring an adequate water supply during a design-basis LOCA. ABB-CE's method for calculating minimum CSS pump NPSH during a design-basis LOCA conforms with RG 1.1 "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," and SRP Section 6.2.2. The calculated minimum CSS pump NPSH with all appropriate conservative as-

sumptions is 63.7 kPa (21.3 ft) for the pumps at their runout flow of 25,000 L/minute (6,500 gpm).

ABB-CE also discussed the design features of the IRWST and holdup volume tank (HVT) which will maintain a sufficient water inventory during all phases of a design-basis LOCA and will prevent debris from entering the CSS pumps. The spillways and screens between the HVT and the IRWST and at the entrance to the suction piping to the CSS pumps prevent particles greater than 2.3 mm (0.09 in.) in diameter from entering the CSS. Using the methods in RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," ABB-CE calculated that no air would enter the CSS pump at the normal rate of pump flow and that a maximum of 2 percent of air would enter at the CSS pump runout flow. These analyses and design evaluations conform to the requirements of RG 1.82 and SRP Section 6.2.2.

In CESSAR-DC Section 6.5, ABB-CE states that each CSS train will have its own spray header and nozzles which are located in the upper region of the containment and are designed to provide maximum effective coverage of the containment volume. The maximum fall height is 25.3 m (83 ft) and the average drop residence time in the containment atmosphere is 13 seconds. The SPRACO 1713A nozzle is used in the CSS and provides a mean drop size of 530 microns and a median drop size of 230 microns for a nozzle pressure differential of 275.8 kPa (40 psid). The results of the containment temperature-pressure analyses for LOCAs and MSLBs in CESSAR-DC Section 6.2 show that the CSS and the passive heat sinks effectively reduce the long-term containment temperature and pressure.

In Amendment U of CESSAR-DC Section 6.2, ABB-CE states that the efficiency of the containment spray for heat removal is a function of droplet size, fall height, and the containment atmosphere steam-to-air ratio. For the System 80+ design, the assumed droplet size is 1000 microns and the fall height is 6.1 m (20 ft), and these are satisfied by the design of spray nozzles. The volume of the containment covered by the sprays does not affect the total heat removal.

On the basis of the in-containment analysis presented in Section 6.2.1 of this report and the staff's review, the design of the containment heat removal system, with suitable redundancy, complies with the requirements of GDC 38 as it relates to containment heat removal capability to reduce rapidly the containment pressure and temperature following any LOCA.

In CESSAR-DC Section 6.2.2.4, ABB-CE states that preoperational tests will be conducted to verify proper

operation of the CSS. The tests include calibration of instrumentation, verification of adequate pump performance, verification of the operability of all associated valves, and verification that the spray headers and spray nozzles are free of obstructions. Preoperational hydrostatic tests will be conducted in accordance with Section III of the ASME Boiler and Pressure Vessel Code. In CESSAR-DC Section 6.5.4.2, ABB-CE states that the CSS is designed and installed to permit inservice inspection (ISI) and testing in accordance with ASME Boiler and Pressure Vessel Code Section XI. The complete schedule for CSS tests and inspections is described in CESSAR-DC Chapter 16. The staff has reviewed the testing and inspection of the containment heat removal system in meeting the testing and inspection requirements of GDC 39 and 40 and finds them acceptable.

CESSAR-DC Section 6.2.2, does not have sufficient information for the staff to adequately review this section to the criteria in SRP Section 6.2.2. However, CESSAR-DC Section 6.5 has more detailed information on the CSS. Therefore, the staff reviewed information from CESSAR-DC Sections 6.5 and 6.2.2 on the CSS as a containment heat removal system.

On the basis of this review, the staff concludes that the containment heat removal system of System 80+ conforms to the guidance in SRP Section 6.2.2, RGs 1.1 and 1.82, and the requirements of GDC 38, 39, and 40 with respect to the adequacy of heat removal capability and CSS design, pump NPSH, availability of recirculation flow from IRWST, redundancy of components, testing, and inspection of the CSS.

6.2.3 Secondary Containment Functional Design

The System 80+ containment design has a secondary containment comprising the containment shield building and the volume that it encloses outside of the primary containment structure. This building is a reinforced-concrete structure in the shape of an up right cylinder with a hemispherical dome that completely encloses the steel sphere of the primary containment. It has an inner radius of about 32 m (105 ft), a height of about 65.5 m (215 ft), and is 0.9 m (3-ft) thick in both the cylinder and dome sections.

The secondary containment shield building shields against radiological effects and missiles and encloses safety-related equipment outside the primary containment structure. The annulus, which is the space between the shield building and the primary containment, helps reduce radioactivity releases to the environment. This annulus consists of 17,500 m³ (617,000 ft³) of net free volume.

An annulus ventilation system (AVS) maintains a negative pressure inside the annulus and vent annulus air through filtered vents which remove airborne radioisotopes. The AVS is actuated by the containment spray signal or by manual means. It consists of two 100-percent-capacity systems of fans, ducts, and filters. Each system is completely independent, having no electrical or mechanical connections to the other which could cause a common-mode failure. The system consists of two ducts, one in the upper annulus and one in the lower, with a fan drawing air from the upper duct through a demister, and through high-efficiency particulate air (HEPA) and carbon filters. The filtered air can be discharged either back to the annulus through the lower duct or out the plant vent. Each fan is powered by a separate Class 1E emergency diesel generator (EDG). In Section 6.2.3.4 of CESSAR-DC, ABB-CE states that tests and inspections will be performed to demonstrate the capability of components, and to demonstrate that the system can perform its function according to the design criteria. These tests include manufacturer testing, operational testing before initial startup, and periodic tests and inspections.

The system is designed to maintain a negative pressure in the annulus of -0.1 kPa (-0.5 in. water gauge) during a LOCA. The AVS is an engineered safety feature (ESF) and, as such, is credited in accident analyses. It is designed to withstand seismic events, tornados, winds, and missile. In the CESSAR-DC, ABB-CE states that components of the AVS are designed, manufactured, and tested in accordance with applicable Institute of Electrical and Electronics Engineers (IEEE) and other relevant standards. Responding to a staff question, ABB-CE committed to following the staff guidance in RG 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," with one exception taken to Position C.5 of this guide. In CESSAR-DC Section 6.2.3.4.1, ABB-CE states that "HEPA filters will be tested for efficiency initially at the factory and the U.S. NRC Quality Assurance Station in accordance with MIL-STD-282." RG 1.52 Position C.5 refers to HEPA testing in accordance with ASME N510, ASME AG-1-1991, and ASTM D 3803-1989. In the DSER, the staff stated that ABB-CE had not explained the difference between these standards and the reason for taking this exception. This was designated as DSER Open Item 6.2.3-1. In Amendment Q, ABB-CE revised CESSAR-DC Section 6.2.3.4.1 to state that the System 80+ design will comply with RG 1.52 without any exception. The staff finds this acceptable. On this basis, DSER Open Item 6.2.3-1 is resolved.

In the DSER, the staff stated that it was not clear that the filter train will include the minimum instrumentation

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requirements listed in SRP Table 6.5.1-1. This was designated as DSER Open Item 6.2.3-2. In Amendment Q, ABB-CE revised its minimum instrumentation requirements listed in CESSAR-DC Table 9.4-3A to conform to the minimum instrument requirements in SRP Table 6.5.1-1. The staff finds this revised table acceptable. On this basis, DSER Open Item 6.2.3-2 is resolved.

In the DSER, the staff stated that the COL applicant should perform local leak rate tests on the AVS in accordance with Appendix J to 10 CFR Part 50. The COL applicant should perform Type B tests for bellows and seals and Type C tests for containment isolation valves (CIVs). This was designated as COL Action Item 6.2.3-1 in the DSER. By letter of December 18, 1992, ABB-CE agreed to this COL action item. However, since ABB-CE has committed to comply with Appendix J in CESSAR-DC Section 6.2.6, the staff concludes that COL applicants will perform the tests required above without additional requirements. Therefore, COL Action Item 6.2.3-1 becomes unnecessary.

ABB-CE analyzed the pressure in the annulus building pressure analysis to determine the time needed to reestablish subatmospheric pressure in the annulus and to demonstrate the adequacy of the AVS design. ABB-CE assumed a design-basis LOCA in the primary containment and analyzed the annulus using the ABB-CE CONTRANS computer code, which solves the conservation of mass and energy equations during each time step, while accounting for all heat sources into and losses out of the annulus.

In modeling this scenario, ABB-CE assumed a minimal AVS flow capacity and maximum initial air temperatures in both the primary and secondary containments. These assumptions would yield the maximum rise in temperature and pressure in the annulus. This analysis included the expansion of the primary containment shell induced by temporal heat and pressure during the LOCA. ABB-CE used natural convection heat transfer in modeling the heat transfer between the annulus air, the outer primary containment sphere, and the inner surface of the shield building. However, ABB-CE failed to include the radiant heat transfer to the secondary containment; this should be included in order to be consistent with Item II.1.a of SRP Section 6.2.3. This was designated as DSER Open Item 6.2.3-3. In Amendment N to CESSAR-DC, ABB-CE revised its assumptions to include the radiant heat transfer. ABB-CE assumed a leakage rate into the annulus of 28.3 m³/per minute (1000 cfm) and set the AVS flow to 453 m³/minute (16,000 cfm) corresponding to only one of the two AVS trains operating. The leakage rate of 28.3 m³/minute (1000 cfm) is much greater than the 0.5-percent-per-day design leak rate of the primary containment.

The results of the annulus analysis are a peak annulus pressure of 2.3 kPa (9.3 in. of water or 0.34 psig) at 63 seconds, which drops to a negative pressure at 104 seconds and a peak temperature of 91 °C (195 °F) after about 11.9 hours. After 11.9 hours, the temperature gradually drops. In the DSER, the staff stated that although the CESSAR-DC includes no design values for annulus pressure and temperature, ABB-CE stated that these peak values are less than the design limits. The staff asked ABB-CE to list the design limits in the CESSAR-DC. This was designated as Confirmatory Item 6.2.3-1 in the DSER. ABB-CE indicated that the design values would not be available until the final structural analysis was preformed. However, the pressurization load has been included in the structural analysis and is insignificant relative to the other structure loads. ABB-CE made the design and analysis calculations available. The staff finds ABB-CE's response acceptable. On this basis, Confirmatory Item 6.2.3-1 is resolved.

ABB-CE also failed to address the issue of primary containment leakage bypassing the secondary containment as specified in SRP Section 6.2.3 (Item II.4.a) and BTP CSB 6-3. This was designated as DSER Open Item 6.2.3-4. In Amendment Q, ABB-CE listed the potential bypass leakage paths in Table 6.2.4-2 of CESSAR-DC and specified the design bypass leak rate of 0.05 percent volume/day in Table 6.2.1-3. In CESSAR-DC Section 6.2.3.4, ABB-CE committed to test the bypass leak paths by local leak tests in accordance with Appendix J to 10 CFR Part 50. On this basis, the staff finds ABB-CE's commitment acceptable; therefore, DSER Open Item 6.2.3-4 is resolved.

On the basis of this review, the staff concludes that the design, analysis, and leak testing of the secondary containment conform to the guidance in SRP Section 6.2.3, and RG 1.52 and conforms to the requirements in GDC 4, 16, and 43, and 10 CFR Part 50, Appendix J with respect to structures, systems, and components (SSC) important to safety being protected against dynamic effects, functional capability of the secondary containment, the analysis of the pressure and temperature responses, periodic functional testing of AVS, and periodic leakage testing of secondary containment bypass leakage.

6.2.4 Containment Isolation System

The containment isolation system (CIS) is designed to four basic tenets: (1) the use of a double barrier for all containment penetrations that are not required to function after a DBA; (2) automatic and leaktight closure of all penetrations which create paths to the containment atmosphere after a DBA; (3) containment penetrating fluid system barrier leak testing methods; and (4) periodic operability testing of CIVs.

In the CESSAR-DC, ABB-CE states that the CIS design conforms to the following regulations and standards:

- GDC 52: to provide the capability for containment leakage rate testing
- GDC 54: to provide leak detection, isolation, and containment capability including redundancy and reliability for the piping systems and related components penetrating the containment
- GDC 55: to provide CIVs for reactor pressure boundary lines penetrating containment
- GDC 56: to provide CIVs for each line that connects directly to the containment atmosphere and penetrates containment
- GDC 57: to provide at least one CIV for each line that penetrates containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere
- ANSI N18.2 and RG 1.29, "Seismic Design Classification:" Design and construction of the CIV and connecting piping (Safety Class 2 and seismic Category I).
- ASME Code: Qualification of the CIV and piping in accordance with Section III of the ASME Code as Class 2 components
- ANSI/American Nuclear Society (ANS) 56.2: CIV maximum allowable actuation times
- 10 CFR Part 50 Appendix J: Containment leak rate testing

ABB-CE also commits to CIS design features which include: missile protection; design pressure and temperature margin; pipe whip and jet impingement protection; LOCA debris particle size limitation in fluid drawn from the containment sump; postaccident environmental qualification of CIVs; and safety function consistent with CIV operators and power sources.

In CESSAR-DC Section 6.2.4, ABB-CE fails to commit to the following standards and regulations specified in SRP Section 6.2.4:

- RG 1.11: instrument line containment isolation
- RG 1.26: quality group classifications
- RG 1.141: containment penetration system classification

- BTP CSB 6-4: containment purging during normal operation

In the DSER, the staff stated that ABB-CE must clarify whether or not the CESSAR-DC will conform to each of the regulations and standards listed above. This was designated as DSER Open Item 6.2.4-1. In Amendments N and Q, ABB-CE committed in CESSAR-DC Sections 6.2.4.1.2 and 9.4.6.1, and Table 6.2.4.1 to complying with these requirements. On this basis, DSER Open Item 6.2.4.1 is resolved.

In CESSAR-DC Table 6.2.4-1, ABB-CE lists all CIVs with the associated location, arrangement, position, actuation, and test information for each valve. Designated as DSER Open Item 6.2.4-2, this table and its accompanying notes and text should include the following information:

- valve closure time
- identification of essential and nonessential systems as described in RG 1.141
- applicable GDC number (such as 55, 56, or 57)
- pipeline size
- distance of isolation valve from the containment
- ESF function

In subsequent amendments, ABB-CE placed the requested information in CESSAR-DC Table 6.2.4-1, with the exception of pipeline sizes and distance of isolation valve from the containment. ABB-CE stated that this level of detailed information was beyond the scope of design certification. However, ABB-CE committed to GDC 54, which requires that CIVs be located as close as practical to the containment. The missing information will be available by the time that COL applicants determine pipe design. Therefore, DSER Open Item 6.2.4-2 becomes COL Action Item 6.2.4-1 and is acceptable.

Recognizing that the aforementioned information was missing, the staff reviewed CESSAR-DC Table 6.2.4-1 in accordance with SRP Section 6.2.4. This review included determining if the valves will be arranged correctly for each type of containment penetration, as specified in GDC 55, 56, or 57. The staff studied the effect of the four possible CIV positions (normal, fail safe, shutdown, and accident) on the function of the specific system during a DBA and determined whether or not the CIV it will be connected to the reactor coolant system (RCS) pressure boundary, the containment atmosphere, or a closed system

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within the containment. The staff also examined the CIV actuation signal and type of actuator to confirm compliance with SRP Section 6.2.4.

In the DSER, the staff designated the following as open items:

- ABB-CE did not specifically commit to RGs 1.11, 1.26, and 1.141, and BTP CSB 6-4, which are cited in SRP Section 6.2.4. On the basis of the preceding evaluation, this open item was closed when Open Item 6.2.4-1 was resolved.
- All CIVs listed in CESSAR-DC Table 6.2.4-1 had no information in the following areas: applicable GDC, line size, the essential or nonessential nature of the system, valve closure times, location of CIV from the containment wall, and a confirmation of whether or not the CIV performs an ESF function. Except for the information to be submitted by COL applicants when the design details of piping and valve locations are available this open item was closed when DSER Open Item 6.2.4-2 was resolved.
- ABB-CE did not indicate that each line directly connecting the containment atmosphere and the environment had a radiation monitor that could isolate the line upon sending a high radiation signal as required in SRP Section 6.2.4 (Item II.m). This was designated as DSER Open Item 6.2.4-3. In Amendment Q, ABB-CE stated in CESSAR-DC Section 6.2.4.3 that the above required dedicated radiation monitors are included in the design. On this basis, DSER Open Item 6.2.4-3 is resolved.
- In CESSAR-DC Section 6.2.4.2, ABB-CE states that CIVs and their operators are designed to withstand a maximum integrated radiation dose of $4.0\text{E-}9$ Gy ($4.0\text{E-}7$ rad) during the life of the plant. This value is not physically reasonable. ABB-CE did not state the type of radiation used in calculating this dose (such as gamma or neutron). This was designated as DSER Open Item 6.2.4-4. In Amendment J, ABB-CE revised CESSAR-DC Section 6.2.4.2 stating that CIVs, operators, and associated instrumentation and controls are environmentally qualified for the normal and postaccident environments in which they are located as discussed in CESSAR-DC Section 3.11. The staff finds this revision acceptable. On this basis, DSER Open Item 6.2.4-4 is resolved.
- ABB-CE had not responded to the staff's request for additional information (RAI) [Q480.37(d)] regarding an analysis of the radiological results of a DBA with the containment purge system initially open. This was

designated as DSER Open Item 6.2.4-5. In response, ABB-CE indicated that the question had been included in a letter from ABB-CE dated December 24, 1991. In the response letter, ABB-CE stated that the requested analysis was included in CESSAR-DC Section 15.6.5, where ABB-CE stated that in the dose analysis, the containment purge is assumed to be terminated 30 seconds after the actuation of containment isolation actuation signal. This is consistent with the valve closure time for the low-volume purge valves specified in CESSAR-DC Table 6.2.4-1. The high-volume containment purge system CIVs are required by Technical Specifications (TSs) to be seal closed when the RCS exceeds hot shutdown conditions. On this basis, DSER Open Item 6.2.4-5 is resolved.

In Amendment U to the CESSAR-DC, ABB-CE took a new position regarding the containment penetrations for the four SIS direct vessel injection (DVI) lines and the two parallel shutdown cooling system (SCS) discharge lines and SIS warmup bypass lines. They pointed out that these penetrations are part of closed loops outside of containment and have only a single outboard containment isolation valve for each line. By taking this position, the inboard check valves next to the containment are no longer considered to be CIVs, and therefore, are not subject to Type C leak testing (see discussion in Section 6.2.6 of this report.) The justifications for the single isolation valve arrangement are provided in CESSAR-DC Section 6.2.4.3. These lines are in an ESF system and will have higher system reliability in its ECCS function by minimizing the number of valves in the line. There are two barriers on each line for containment isolation; one is the outboard containment isolation valve, and the other is the closed loop piping outside containment.

The staff has reviewed, according to SRP Section 6.2.4, the proposal that considers the closed loop outside containment as the second isolation barrier. The closed loop SIS outside containment is protected from missiles, is designed to seismic Category I and Safety Class 2, and has a design temperature and pressure exceeding the containment design temperature and pressure. The integrity of the closed system is maintained during normal plant operation and is periodically pressure tested in accordance with the pre-service and in-service inspection programs required by 10 CFR 50.55a and the ASME Code, as described in CESSAR-DC Section 6.6. The integrity of the system is checked during in-service testing of safety related pumps and valves, as described in CESSAR-DC Section 3.9.6. Based on the above review, the staff concurs with ABB-CE that the closed loop outside containment can be considered as the second barrier for the containment isolation. Therefore, the proposed approach using a single isolation valve for each of these SIS lines is acceptable.

The staff has reviewed the CIS in accordance with SRP 6.2.4 and BTP CSB 6-4. On the basis of its review, the staff concludes that the CIS design complies with the requirements of GDC 1, 2, 4, 16, 54, 55, 56, and 57 and Appendix K to 10 CFR Part 50, as related to containment isolation, with one exception. The exception has been designated as COL Action Item 6.2.4-1, which requires COL applicants to submit information about pipe sizes and distance of isolation valves from containment.

6.2.5 Containment Combustible Gas Control

The System 80+ combustible gas control design consists of two separate systems: the containment hydrogen recombiner system (CHRS) and the hydrogen mitigation system (HMS). The CHRS will prevent hydrogen combustion during and after a design-basis LOCA by ensuring that the hydrogen concentration in the containment never reaches 4 percent, its lower flammability limit. The HMS will keep the containment hydrogen concentration below 10 percent to preclude detonations during and after a severe accident.

Containment Hydrogen Recombiner System

The CHRS consists of two fully redundant 100-percent capacity and independent loops each with a hydrogen recombiner, piping connecting to the containment atmosphere, instrumentation, and an electrical power supply. CHRS components are designed to withstand LOCA containment pressure and temperature in accordance with the standards for seismic Category I, Safety Class 2 systems specified in the ASME Boiler and Pressure Vessel Code Section III. These components are protected from missiles, jet impingement, or pipe ruptures. The CHRS is also designed in accordance with guidance from Section 4 of IEEE 279-1971 and the requirements of IEEE 308-1972. The CHRS also include provisions to remove IRWST sump radiolysis hydrogen and eliminate any possible hydrogen pockets within the containment volume. Class 1E EDG power is available in case of a loss-of-offsite power (LOOP). The CHRS has redundant hydrogen analyzers at the inlet and outlet of the hydrogen recombiner.

The CHRS starts within about 84 hours after a design-basis LOCA. ABB-CE presented the results of a failure-modes-and-effects analysis of the CHRS. This analysis shows that a redundant CHRS train and a redundant hydrogen analyzer provide sufficient protection from any postulated single failure.

ABB-CE performed a conservative analysis of the maximum hydrogen concentration present in the containment following a design-basis LOCA. Using the guidance in RG 1.7 and TID-14844 fission product activity release, ABB-CE calculated the hydrogen production rate as a function of time after a LOCA. ABB-CE analyzed core solution radiolysis, sump solution radiolysis, containment material (i.e., zinc and aluminum) corrosion, core zirconium-water reaction, and reactor coolant dissolved hydrogen. The results of this calculation are graphically presented in the CESSAR-DC for both the containment atmosphere and the IRWST regions with and without the startup of one of the two recombiners at 84 hours. For both regions, these results show that the CHRS design maintains a hydrogen concentration below 4 percent within the containment.

The staff reviewed the CHRS for design-basis LOCAs and found that the design conformed to the regulations and standards in SRP Section 6.2.5. ABB-CE calculated the design-basis LOCA hydrogen production rate in accordance with RG 1.7. The CHRS design meets the standards for performance, independence, redundancy, inspection, and quality as given in GDC 5, 41, 42, and 43. ABB-CE's analysis of design-basis LOCA containment hydrogen concentration with conservatively calculated hydrogen production and a single active failure of the CHRS shows that a startup of the CHRS at 84 hours into the event ensures that the hydrogen concentration will not reach its 4-percent flammability limit.

Hydrogen Mitigation System

The HMS consists of two independent 100-percent capacity redundant groups of hydrogen igniters which can be powered from EDGs, the gas turbine, or Class 1E batteries through a dc-ac inverter in case of a LOOP. The HMS is not nuclear safety grade, but is designed to withstand seismic Category I and other severe accident environmental conditions. Igniters are positioned within the containment where local pockets of hydrogen may collect during a severe accident. Each HMS igniter is an ac glow plug integrated with its own stepdown transformer in a water-tight enclosure that meets National Electrical Manufacturers Association Type 4 specifications.

The HMS is manually started from the control room to accommodate the hydrogen produced by a reaction of 100 percent of fuel-clad metal with the coolant water as defined in 10 CFR 50.34(f). The igniters burn hydrogen without endangering critical equipment inside the containment and maintain a hydrogen concentration below 10 percent during a postulated severe accident. The staff further discusses its

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review of the adequacy of HMS in Section 19.2.3.3.1 of this report.

6.2.6 Containment Leakage Testing

ABB-CE committed to containment leakage testing for the System 80+ plant in accordance with Appendix J to 10 CFR Part 50 with the following exceptions:

- (1) The COL applicant may use the mass point leak rate test method in ANSI/ANS 56.8-1987 as an alternative to Type A testing method specified in ANSI 45.4-1972.
- (2) Leaks occurring during the Type A test that could affect the test results will not prevent completion of this test if: (a) the leaks are isolated for the balance of the test; (b) the leaking component had a "pre-maintenance" local leak rate test whose results, when added to those from the Type A test, are in conformance with the acceptance criteria of Appendix J; or (c) a "postmaintenance" local leak rate test of the leaking component(s) is performed and the results, when added to those from the Type A test, conform to the acceptance criteria of Appendix J.

The first exception is acceptable because the current version of Section III.A.3 of Appendix J to 10 CFR Part 50 includes the ANSI/ANS 56.8-1987 method (mass point method) as an acceptable alternative.

The second exception does not conform to the requirements of Appendix J to 10 CFR Part 50. Section III.A.1.(a) of Appendix J requires that a Type A test, defined as a test to measure the primary containment overall integrated leakage rate be terminated if, during this test, potentially excessive leakage paths are identified which would either interface with satisfactory completion of the test or which would result in the Type A tests not meeting the applicable acceptance criteria of Section III.A.4(b) or III.A.5(b). Section III.A.1(a) further requires that, after terminating a Type A test due to potentially excessive leakage, the leakage through the potentially excessive leakage paths be measured using local leakage testing methods and repairs and/or adjustments to the affected equipment be made. The Type A test shall then be conducted. ABB-CE proposes that, when leakage is found during a Type A test, the test not be terminated. Instead, ABB-CE proposes that leaks be isolated and the Type A test continued. After completion of the modified Type A test (i.e., a Type A test with the leakage paths isolated), local leakage rates of those paths isolated during the modified Type A test will be measured before or after the maintenance to those paths.

ABB-CE proposes that the adjusted "as-found" leakage rate for the Type A test be determined by adding the local leakage rates measured before maintenance to those previously isolated leakage paths, to the containment integrated leakage rate determined in the modified Type A test. This adjusted "as-found" leakage rate is to be used in determining the scheduling of the periodic Type A tests in accordance with Section III.A.6 of Appendix J.

Finally, ABB-CE proposes that the acceptability of the modified Type A test be determined by calculating the adjusted "as-left" containment overall integrated leakage rate and comparing this to the acceptance criteria of Appendix J. The adjusted "as-left" Type A leakage rate is determined by adding the local leakage rates measured after any maintenance to those previously isolated leakage paths, to the leakage rate determined in the modified Type A test.

The differences between the proposed leak testing and the requirements in Section III.A.1(a) of Appendix J are that: (1) the potentially excessive leakage paths will be repaired and/or adjusted after completion of the Type A test rather than before the test; and (2) the Type A test leakage rate is partially determined by calculation rather than by direct measurement. With respect to the first issue, the staff does not identify any significant difference in the end result (i.e., the "as-left" local leakage rates will be maintained within an acceptable range). With respect to the second issue, the measured "as-left" local leakage rates will represent a relatively small correction to the containment overall integrated leakage rate measured in the modified Type A test. Accordingly, there will be insignificant difference between the calculated "as-left" containment leakage rate (i.e., a modified Type A test) and one which would be directly measured in compliance with the requirements of Section III.A.1.(a).

The staff concludes that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose because ABB-CE has proposed acceptable alternatives that accomplish the intent of the regulation. On this basis, the staff concludes that a partial exemption from the requirements of Section III.A.1.(a) of Appendix J is justified.

In CESSAR-DC Section 6.2.6 and Table 6.2.4-1, ABB-CE presents information on the System 80+ containment leakage testing program, including the planned leak test data for specific CIVs. In CESSAR-DC Table 6.2.4-1, ABB-CE lists those CIVs which are vented and drained for the Type A test and those CIVs which are subject to the Type C test, and justifies those CIVs not included in the

Type C test program. ABB-CE presents the following justifications for not performing CIV Type C tests:

1. CIVs on piping connected to the secondary side of the steam generator would leak into the containment because, during a design-basis LOCA, the secondary side pressure is higher than the primary-side pressure.
2. The water always present in the IRWST seals CIVs on piping connected directly to the IRWST.
3. The discharge pressure from the safety injection pump effectively seals against leakage for CIVs on pump discharge (or injection) lines.
4. The system SCS with these CIVs must maintain safe shutdown conditions that cannot be tested without compromising safety, but that will be separately water tested as part of the RCS pressure boundary.

On the basis of SRP Section 6.2.6, the staff concluded in the DSER that justification 2 is acceptable for excluding these CIVs from Type C leak rate testing. However, the staff found in the DSER that justification 1 was not acceptable. If steam generator tubes leak, the secondary-side pressure would not be higher than that on the primary side, and CIVs on the secondary side would be the only isolation barrier for the containment penetration lines. In addition, the recent issue of the steam generator tube rupture bypass leakage prompted the concern of the leak testing of those valves. Therefore, the staff stated in the DSER that these valves should be leak tested with operating fluid. This was designated as DSER Open Item 6.2.6-1.

After the staff issued the DSER, ABB-CE stated that currently operating plants do not require that of CIVs on the secondary side of steam generators be leak tested and that the probability of steam generator tube rupture for the System 80+ design is lower than the probability for currently operating plants because of the change of material in the steam generator tubes from Inconel 600 to Inconel 690. Furthermore, both the radiological and environmental consequences of the leakage from those valves are bounded by the existing safety analysis (CESSAR-DC Chapter 15) and equipment qualification program (CESSAR-DC Chapter 3). To further reduce the risk of containment bypass due to steam generator tube rupture events, ABB-CE committed in a letter dated October 6, 1993, to add N-16 radiation monitors to provide early detection of steam generator tube rupture, thereby reducing the probability for containment bypass from opening of the main steam safety valves during a steam generator tube rupture event. In addition, ABB-CE stated in the letter that a design change to the System 80+ component cooling water system would assure continued operation of the

steam turbine bypass system following a safety injection actuation signal (SIAS).

ABB-CE revised CESSAR-DC Table 3.9-15 to add leak testing of these valves for gross leakage in inservice leakage testing. In addition, ABB-CE states that these valves are subject to the valve inspection programs described in CESSAR-DC Section 3.9.6.2. On the basis of these justifications and additional mitigation measures committed to by ABB-CE, the staff concurs with the position of not requiring Type C leak testing on those valves; therefore, DSER Open Item 6.2.6-1 is resolved.

The staff did not find justifications 3 and 4 acceptable in the DSER because multiple systems would allow the CIVs on one loop to be tested while the others are available. The two 100-percent redundant SCS would ensure safe shutdown with one system operating while the CIVs in the other are being leak tested. If the safety injection pump fails and the system switches from cold-leg to hot-leg injection, any leakage from the system safety injection pump CIVs would pass to the environment. Therefore, in the DSER, the staff concluded that both the SCS and safety injection pump system CIVs should be tested for leaks in accordance with Appendix J to 10 CFR Part 50. This was designated as DSER Open Item 6.2.6-2.

In Amendment U to the CESSAR-DC, ABB-CE rearranged valve elevation so that SIS valves SI-602, 603, 616, 626, 636, and 646 are approximately 1.2 m (4 ft) below the minimum IRWST water level and SCS valves SI-600 and 601 are approximately 0.44 m (1.5 ft) below the minimum water level. The minimum IRWST water level is at elevation 24.5m (80.5 ft) which is determined by the calculated minimum IRWST water level following a large LOCA. By using this valve re-arrangement, the IRWST will provide a manometer effect to establish a water seal at the valves because the containment pressure is exerted on the surface of the IRWST liquid and the SIS forms a closed loop with containment following a pipe break. ABB-CE states that it complies with the intent of the regulation in 10 CFR Part 50, Appendix J in maintaining water-sealed valves.

The staff has reviewed the proposed alternative. Appendix J to 10 CFR Part 50, Section III.C.3(b) states that the installed isolation valve seal water system fluid inventory is sufficient to assure the sealing function for at least 30 days at a pressure of 1.1 Pa. The proposed design of water-sealed isolation valves conforms to the requirement of 30-day water inventory but not on the sealing pressure of 1.1 Pa. However, the staff finds that the closed loop and the manometer effect provide sufficient water sealing as long as the integrity of the closed loop and the elevation differential between the valves the water level are main-

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tained. As a result of the review, the applicant has committed to provide: (1) periodic pressure testing as described in CESSAR-DC Sections 3.9.6 and 6.6 to ensure the integrity of the closed loop SIS outside containment is being maintained; and (2) a pre-operational test as described in CESSAR-DC Section 14.2 to ensure the existence of the water seal. Based on the staff review and ABB-CE's commitment to the above periodic and pre-operational tests, the staff concludes that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose because ABB-CE has proposed acceptable alternatives that accomplish the intent of the regulation. On this basis, the staff concludes that a partial exemption from the requirements of Section III.C.3(b) is justified because the alternative water-sealed-valve design accomplishes the objectives of the regulatory requirement of sealing pressure of 1.1 Pa.

In addition, ABB-CE states that the check valves SI-113, 123, 133, 143 inside containment are not considered as CIVs, and therefore, are not included in Type C leak testing. However, these check valves are included in the inservice testing (IST) Program as described in CESSAR-DC Section 3.9.6.2. As discussed in Section 6.2.4 of this report, the staff finds ABB-CE's justification for a single outboard containment isolation valve for each of the SIS lines to be acceptable. Therefore, excluding these check valves from the requirement of Type C leak testing is acceptable. Based on the above evaluation, DSER Open Item 6.2.6-2 is resolved.

In accordance with 10 CFR Part 50 (Appendix J Section IV.B) the staff states in SRP Sections 6.2.3 and 6.2.6 that applicants for dual-containment nuclear plants or plants with secondary containments must verify that designed leakage limits are not exceeded. In the DSER, the staff stated that ABB-CE had not committed to performing any leak rate tests for the annulus region between the primary containment sphere and the secondary containment shield building. Therefore, the lack of a leak rate test for the System 80+ secondary containment was not acceptable. This was designated as DSER Open Item 6.2.6-3. In Amendment Q, ABB-CE committed in CESSAR-DC Section 6.2.6.5 that the inleakage to the reactor building is checked preoperationally and periodically by a negative pressure test to verify that the AVS achieves a negative pressure of -125 Pa (-0.5 in. water gauge) within 110 seconds of actuation and that the potential bypass leakage would be leak tested in accordance with 10 CFR Part 50, Appendix J. The sum of the results of potential bypass leak path tests must be proved to be lower than 0.05 volume percent per day. The staff finds ABB-CE's commitment acceptable. On this basis, DSER Open Item 6.2.6-3 is resolved.

On the basis of its review, the staff concludes that the System 80+ containment leakage testing program in the CESSAR-DC is acceptable, and complies with the guidance in SRP Section 6.2.6 and the requirements in Appendix J to 10 CFR Part 50 and GDC 52, 53, and 54 as related to containment leak testing with the exceptions identified by ABB-CE that have been found acceptable by the staff.

6.3 Safety Injection System

The SIS will provide core cooling and additional shutdown capability for accidents that result in significant depressurization of the RCS. These accidents include the failure of the RCS piping up to, and including the double-ended break of the largest pipe, breaks in the main steam piping, a control element assembly ejection accident, and a steam generator tube rupture event. The staff reviewed the SIS described in CESSAR-DC Section 6.3, for each of the areas listed in paragraph I, "Areas of Review," of SRP Section 6.3 in accordance with the guidelines provided in the SRP review procedures. The staff reviewed this system for conformance with the following requirements:

- GDC 5 as it relates to SSCs important to safety shall not be shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety functions.
- GDC 17 as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded and that the core is cooled during anticipated operational occurrences and accident conditions.
- GDC 19 as it relates to the control room design from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions.
- GDC 20 as it relates to the design of the protection system having functions to initiate automatically the operation of appropriate systems to assure that specified acceptable fuel design limit are not exceeded as a result of anticipated operational occurrences.
- GDC 35, 36, and 37 as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection

of important components, and to permit appropriate periodic pressure and functional testing.

- GDC 56 as it relates to the design of containment isolation valves to satisfy specific acceptance criteria.
- 10 CFR 50.46 and Appendix K to 10 CFR Part 50 as they relate to the ECCS being designed so that its cooling performance is in accordance with an acceptable evaluation model.

The staff reviewed the system description, piping and instrumentation drawings, and the responses to the staff requests for additional information (RAI-440 series) to verify that the SIS will provide abundant core cooling during the injection phase, with and without offsite power after a single failure. CESSAR-DC Section 6.3.3 describes the LOCA analyses and compliance with 10 CFR 50.46, Appendix K to 10 CFR Part 50, and GDC 35. The results of the staff's review of these analyses are included in Section 15.3.7 of this report.

6.3.1 System Design and Functional Requirements

The SIS consists of active and passive injection systems. The active components of the SIS are found in four mechanically separated trains, each consisting of a safety injection (SI) pump and associated valves. Each SI pump has its own suction line from the IRWST and its own discharge line to a DVI nozzle on the reactor vessel. The SI pumps are actuated by a SIAS at 11,000 kPa (1,600 psia) during a LOCA, a steamline break, a steam generator tube rupture, or a control element assembly ejection event. The passive components of the SIS are four identical pressurized safety injection tanks (SITs). The SITs contain borated water covered by nitrogen pressurized to at least 4,100 kPa (600 psia). When the RCS pressure falls below the tank pressure, borated water is forced from the SITs into the reactor vessel through DVI nozzles.

Long term cooling (LTC) for large-break LOCAs is accomplished by manually opening RCS hot-leg injection valves for SI pumps 3 and 4, and manually closing the corresponding DVI flowpath valves. The DVI nozzle flowpaths of SI pumps 1 and 2 will remain open. This configuration with SI pumps 3 and 4 injecting to the hot legs and SI pumps 1 and 2 injecting into the respective DVI nozzles cools the core for the large-break LOCAs. The SIS is designed so that one of the hot-leg injection systems and one of the DVI injection systems remains functional during the worst single failure, which is the failure of one of two EDGs to start.

The SIS is able to cool the reactor core and supplies additional shutdown capability. ABB-CE selected the SIS

functional parameters so that the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50 are for a LOCA with concurrent LOOP and a single-failure event. The staff reviewed ABB-CE's LOCA analyses, as discussed in Section 15.3.7 of this report. Table 6.2 lists the SIS equipment for the System 80 and System 80+ designs. The System 80 SIS has two high-pressure SI pumps and two low-pressure SI pumps; the System 80+ SIS has four high-pressure SI pumps and no low-pressure SI pump. The SIS injection locations are in the cold legs for the System 80 design. DVI nozzles on the reactor vessel are used for injection for the System 80+ design. The SIS injects borated water from the IRWST with capacity of 2.07×10^6 L (545,800 gal); the System 80 refueling water storage tank has a capacity of 1.90×10^6 L (502,760 gal). The System 80+ design has two more high-pressure safety injection (HPSI) pumps and two less low-pressure safety injection (LPSI) pumps than the System 80 design. In the response to DSER Open Item 6.3.1-1 regarding the SI flowrate at the low-pressure range, ABB-CE indicated in a submittal of November 24, 1992, that the addition of two HPSI pumps supplies the required amount of SI flow delivered at low pressure to compensate for the absence of two LPSI pumps. The emergency core cooling system (ECCS) design for System 80+ is consistent with the EPRI ALWR Utility Requirements Document (URD) requirement, Number 5.4.3.1.2, which states that separate high- and low-head pumps shall not be used for ECCS. For System 80+, the analysis in CESSAR-DC Section 6.3.3 shows that the SIS design can meet the performance acceptance criteria specified in 10 CFR 50.46 for the ECCS. In addition, two shutdown cooling pumps can be manually actuated for injection at low pressure to mitigate consequences of beyond-design-basis-events. On the basis of the staff evaluation discussed in Section 15.3.7 of this report for the LOCA analyses and ABB-CE's guidance to use the SCS in the Functional Recovery Guidelines (FRG), IC-2 of EOGs for injection mode to mitigate beyond-design-basis events, the staff concludes that the SIS design is adequate to address the ECCS acceptance criteria of 10 CFR 50.46 and the RCS inventory control for accident mitigation. On this basis, DSER Open Item 6.3.1-1 is resolved.

In the design of the SIS, ABB-CE considered the reliability of the system (such as the design to conform to seismic Category I requirements) and redundant subsystems (such as the design to meet single-failure criteria) to enhance the overall reliability of the SIS.

The SIS will be initiated either manually or automatically on low pressurizer pressure or high containment pressure. This design feature conforms to the requirements of GDC 20.

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Table 6.2 Safety injection system equipment comparison

	System 80	System 80 +
High Pressure Safety Injection Pumps Nos.	2	4
Design Flow -L per min (gpm)	3090 (815)	3090 (815)
Design Head -m (ft)	869 (2,850)	869 (2,850)
Low Pressure Safety Injection Pumps Nos.	2	0
Design Flow -L per min (gpm)	15,900 (4,200)	
Design Head -m (ft)	102 (335)	
Safety Injection Tank Nos.	4	4
Design Pressure-kPa (psig)	4800 (700)	4800 (700)
Water Volume, Normal -m ³ (cft ³)	52.6 (1,858)	52.6 (1,858)
Refuel Water Tank	1	1
Minimum Water Volume -L (gal)	1.90x10 ⁶ (502,760)	2.07x10 ⁶ (545,800)
Safety Injection Discharge Lines	cold legs	DVI Nozzles on the Reactor Vessel

The SIS can be manually actuated, monitored, and controlled from the control room. This design feature conforms to the requirements of GDC 19. The SIS will be supplemented by instrumentation which enables the operator to monitor and control the SIS equipment after a LOCA so that adequate core cooling may be maintained. The supplemental instrumentation includes the saturation margin monitor, core exit thermocouples, and the reactor vessel level monitoring system. The acceptability of the SIS instrumentation and controls is addressed further in Section 7.3 of this report.

During normal plant operation, the SIS lines (RAI Q440.29) are maintained in a filled condition to minimize the likelihood of water hammer during system initiation.

Since no SIS components are shared between units, this design feature compiles with the requirements of GDC 5.

Containment isolation features for all SIS lines, including instrument lines and compliance with the requirements of GDC 56 and RG 1.11 are discussed in Section 6.2.4. of this chapter.

6.3.2 System Isolation and Overpressurization Protection Design

The design pressure of each RCS DVI line from the reactor vessel up to and including a motor-operated isolation valve outside the containment is equal to RCS design pressure. Each vessel injection line contains three check valves in series inside the containment and a remotely actuated motor-operated valve (MOV) outside the containment. The position of the MOV is indicated in the control room, and this valve can be operated from the control room.

The design pressure of each RCS hot-leg injection line from the RCS up to and including a MOV outside the containment is equal to RCS design pressure. Each hot-leg injection line contains two check valves in series inside the containment and an MOV outside the containment. The position of the MOV is indicated in the control room, and this valve can be operated from the control room.

The SITs have normally open isolation valves in the discharge lines. These valves will have power removed from the motor-operator to preclude undetected closure

during normal operation and inadvertent closure during the SIS injection phase. Each SIT discharge line contains two check valves and a MOV. The design pressure of the pipe is equal to RCS pressure up to and including the second check valve. If leakage into the SITs causes coolant level to reach the setpoints, sensors will activate the SIT high water level and pressure alarms.

The COL holder will periodically test each valve for leaks as specified in the requirements for ISI in Section XI of the ASME Code. At least two check valves are installed in series in the piping between the RCS and low-pressure systems, the design will require periodic leak testing on the check valves, and the MOVs in discharge lines have position indications in the control room and capability to be operated in the control room. Therefore, the staff concludes that the piping arrangement is consistent with the guidelines for isolation specified in SRP Section 6.3, Item III.11.

However, the Commission approved an interfacing-systems loss-of-coolant accident (ISLOCA) position in an NRC letter dated June 26, 1990, "SECY-90-016 Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," requires that the evolutionary ALWR designs (including the System 80+) reduce the possibility of a LOCA outside the containment by designing to the extent practicable all systems and subsystems connected to the RCS to withstand the full RCS pressure. In response to DSER Open Item 6.3.2-1, ABB-CE provided its evaluation of various interfacing systems (including the SIS design) and concluded that the design of the SIS interfacing with the RCS will be in compliance with the requirements specified in SECY-90-016 as discussed for ISLOCA protection. The staff has reviewed the SIS design for meeting the intersystem LOCA requirements. Based on the staff's evaluation in Chapter 20 of this report on Generic Safety Issues-105, the staff finds that intersystem LOCA requirements were adequately addressed by ABB-CE for the SIS design, and the staff concludes that the SIS design is acceptable. On this basis, DSER Open Item 6.3.2-1 is resolved.

6.3.3 Pump Protection Design Requirements

The System 80+ design contains four independent active SIS trains. Each train consists of an SI pump and its associated valves. System reliability is achieved with two electrical divisions, each division supplying power to two SI pumps, associated valves, and supported systems. At least two trains of the injection system would still be activated if a single failure occurred in the power supply

system. In CESSAR-DC Table 6.3.2-2, ABB-CE described a failure modes and effect analysis, which demonstrates that no single active or passive failure could prevent the SIS from fulfilling its short- and long-term functions. Since the design has both the onsite electric power supply system and the offsite electric power supply system, either of which permits the SIS to function, the SIS design complies with the requirements of GDC 17.

The four SI pumps are of sufficient size so that one SI pump, together with the SITs, provides sufficient injection flow to prevent the core from being uncovered for small breaks up to at least a 21.59 cm (8.5 in.)-diameter break size (DVI nozzle). During a large-break LOCA, two SI pumps, together with the SITs, provides the required injection flow to meet functional requirements.

In response to RAI Q440.63, ABB-CE performed an analysis to determine the NPSH available from the IRWST to the SI pumps. ABB-CE performed this analysis with sufficient margin to conform to the regulatory position stated in RG 1.1. ABB-CE committed to verify that the required NPSH for SI pumps is met during the pump procurement process, and included an item to verify available pump NPSH in its inspections, tests, and acceptance criteria (ITAACs) program for the SIS system.

In response to the staff's question (RAI Q440.61) regarding the capability of the SI pump to operate for an extended period of time, ABB-CE state that the SI pump design is similar to the design of boiler feed pumps used in fossil plants. ABB-CE evaluated feedwater pump operating data and indicated that those pumps could be operated without overhaul for more than 5 years. ABB-CE also stated that the pump will be inspected, and submitted a recommendation for parts that should be replaced periodically. The COL applicant holder will perform the routine ISI defined in the TS to verify that the performance of the SI pump meets the functional requirements.

To prevent the SI pumps from runout because of low RCS pressure after a large-break LOCA, the maximum SI flow (RAI Q440.62) is limited to an acceptable runout value when the SIS is set up during preoperational testing with the RCS at atmospheric pressure. The main control room (MCR) includes pressure indicators for the SI lines, flow meters for the SI pump discharge flow, and the associated alarms. These devices will alert the operator to the possible degradation of SIS pump performance. The instrumentation for monitoring the SIS pump performance is operable with and without offsite power.

In response to the staff's questions (RAI Q440.71, 72, and 73) regarding the SI operation at low-flow conditions as described in NRC Bulletin 88-04, ABB-CE stated that the

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arrangement of the SIS mini-flow recirculation lines precludes pump deadhead operation resulting from pump-to-pump interactions as discussed in NRC Bulletin 88-04. The only cross-connection in the SIS arrangement ties two discharge (mini-flow) recirculation lines together downstream of an orifice and a check valve, creating a flow path to the vented IRWST. This piping arrangement will not result in the operation of one SI pump to cause other SI pumps to operate at a flow rate that is lower than the required minimum for pump protection. ABB-CE also confirmed with information from pump manufacturers and utilities, that the minimum recirculation flow previously established for SI pumps applies to the System 80+ design and is adequate to protect pumps at low flow conditions. The design minimum SI pump flow ranges from 322 to 397 L/minutes (85 to 105 gpm) (9.7 percent to 12 percent of best efficiency flow). The staff agrees that the proposed SIS arrangement will avoid pump deadhead operation. However, the staff required that ABB-CE discuss design criteria for the minimum SI recirculation flow, and submit pump operating data or test results demonstrating that the SI pump would be operable for an extended time at the proposed flow range. ABB-CE should also demonstrate in completing the ITAACs that the mini-flow bypass has adequate capability. This was designated as DSER Open Item 6.3.3-1.

Subsequently, in Amendment V to CESSAR-DC, ABB-CE added the following statements to CESSAR-DC Section 6.3.4.1.1:

The pump vendor will perform a pump type test to generate data that verifies the SI pump's ability to operate within the pump design limits at minimum flow for eight hours. Eight hours was selected as a representative time for a natural circulation cooldown for which long term SI pump operation in minimum flow may be required. The pump type test will be performed with fluid conditions representative of those during a natural circulation cooldown for the System 80+ design. The pump type test will monitor the normal pump qualification test parameters and will ensure that the pump bearing oil equilibrium temperature, pump shaft seal leakage temperature, and pump bearing housing vibration are within the acceptance criteria as described in CESSAR-DC Section 6.3.4.1.1. The duration of the test is either eight hours or until the bearing oil temperature, pump shaft seal leakage temperature, and pump bearing housing vibration stabilize, whichever is longer. After completion of the test, the SI pump will be disassembled and the internals inspected with the conditions of the parts

being evaluated for excessive wear as a result of degradation during the pump test.

It should be noted that ABB-CE also stated in CESSAR-DC Section 3.9.6.1 that the System 80+ piping system design can accommodate IST at a flow rate at least as large as the maximum design flow for the pump. In addition, the flow rate through the minimum recirculation flow path can also be periodically measured to verify that it is in accordance with the design specification. The staff finds that the above commitments incorporated into Amendment V to CESSAR-DC provide a reasonable assurance that the SI pump would be operable for the required minimum flow operation condition. On this basis, DSER Open Item 6.3.3-1 is resolved.

6.3.4 Injection Borated Water Source

The borated water to be injected through the SIS is stored in the IRWST inside the spherical steel containment vessel of the reactor building. In the event of a LOCA, the reactor coolant will be collected in the HVT, which will spill the reactor coolant to the IRWST, thus establishing a continuous source of the coolant for the SIS pumps. This design eliminates the need in conventional design to switch from the RWST to the containment sump for LTC.

6.3.5 Evaluation of Single Failures

Following the guidelines in SRP Section 6.3, Item II, the staff reviewed the system description and the piping and instrumentation diagrams to verify that the design will have sufficient core cooling after a LOCA, with and without offsite power, if a single failure occurs.

If offsite power is lost, the worst single failure to affect SIS performance is the failure of one EDG. If one EDG fails, two SI pumps and four SITs are still available for operation. This is more than enough to maintain the SIS flow required after a LOCA, as discussed in Section 15.3.7 of this report.

The staff reviewed the plant's capability for hot-leg injection during the LTC to prevent the accumulation of an excessive concentration of boron in the reactor vessel. The staff finds that one of the hot-leg injection systems and one of the DVI injection systems will remain functional during the worst single failure. The staff concludes that the design of the injection lines and pumps includes sufficient redundancy to ensure adequate hot-leg injection. This design of the hot-leg injection meets the LTC requirements of SRP Section 6.3, Item III.6. The staff docu-

ments its evaluation of the LTC analysis in Section 15.3.8 of this report.

6.3.6 Design Qualification of the SIS

The SIS is designed to seismic Category I requirements in compliance with RG 1.29, "Seismic Design Classification". The structure in which the SIS is housed is designed to withstand a safe-shutdown earthquake and other natural phenomena as discussed in Chapter 3 of this report. In Chapter 3, the staff also addresses the SIS protection against missiles and pipe whip inside and outside the containment.

6.3.7 Testing

In CESSAR-DC, Section 6.3.4, ABB-CE describes the testing requirements for the SIS. The SIS is designed and will be installed to permit the SIS pumps to be tested at full-flow conditions with the reactor at power, which is consistent with the EPRI URD requirements. The COL applicant will perform preoperational testing of the SIS to verify that the system and components perform according to design criteria. The COL applicant will periodically test and inspect the SIS components and subsystems in accordance with the ASME Code, Section XI, which constitutes compliance with GDC 36, to ensure the SIS will operate properly in the event of an accident. The COL applicant will also test such SIS components as pumps and automatic valves to confirm acceptable performance of each active component in the SIS.

In CESSAR-DC, ABB-CE commits to demonstrate the operability of the SIS by subjecting all components to preoperational and periodic testing to be consistent with RG 1.68, "Preoperational and Initial Startup Test Programs for Water Cooled Power Reactors"; RG 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurizer Water Reactors"; and GDC 37, "Testing of Emergency Core Cooling System." SIS testing is COL Action Item 6.3.7-1.

6.3.8 Conclusions

The SIS includes the piping, valves, pumps, instrumentation, and controls for transporting heat from the reactor core after a LOCA, including a pipe break, or safety valve opening in the RCS that could discharge a volume of coolant greater than that to be replaced by the normal makeup system. The staff reviewed the SIS, including piping and instrumentation diagrams, failure modes and effects analyses, and design specifications for essential components. The staff reviewed the CESSAR-DC design

criteria and design bases for the SIS and the manner in which the design conforms to these criteria and bases.

The staff concludes that, the design of the System 80+ SIS is acceptable because it meets the requirements in GDC 5, 17, 19, 20, 36, and 37, and the guidelines in SRP Section 6.3. In Section 15.3.7, the staff documents its evaluation regarding the acceptance criteria of 10 CFR 50.46 and GDC 35.

ABB-CE has included a preoperational testing program in CESSAR-DC Section 14.2.12.1.22 and an ITAAC program that includes testing to (1) verify that the design SI flow and head meet SI pump functional requirements; (2) confirm that the required pump NPSH for the SI pumps is available; and (3) determine the SI runout flow to be used at low-pressure conditions. The staff reviewed the ITAAC for the SIS and finds that the Design Description and ITAAC for the SIS system include the appropriate design commitments which are to be verified and, therefore, determines that they are acceptable. On this basis, DSER Confirmatory Item 6.3.8-1 is resolved.

6.4 Control Room Habitability Systems

The staff reviewed the control room habitability systems in accordance with SRP Section 6.4 to verify that they will conform to the acceptance criteria in the applicable regulations of 10 CFR Part 50. Specifically, the SRP acceptance criteria require the system design to comply with (1) GDC 4, "Environmental and Dynamic Effects Design Bases," regarding accommodating the effects of and being compatible with postulated accidents, including the effects of the release of toxic gases; (2) GDC 19, "Control Room," regarding maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases; (3) Three Mile Island (TMI) requirement 10 CFR 50.34(f)(2)(xxviii), as it relates to evaluating potential pathways for radioactivity and radiation that may lead to control room habitability problems; and (4) TMI Action Plan Item III.D.3.4 (NUREG-0737) requirements as they relate to providing protection against the effects of release of toxic substances, either on or off the site.

Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multiunit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with

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respect to the capability of shared SSCs to perform their required safety functions. This was designated as COL Action Item 6.4-1 in the DSER. Upon further review, the staff has determined that the design described in CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the control room habitability systems comply with the requirements of GDC 5, and COL Action Item 6.4-1 is not required.

During normal and postulated accident conditions, the systems will provide (1) controlled environment for personnel comfort and equipment operability, (2) radiation shielding against airborne radioactivity releases outside the control building through filtration, (3) protection against toxic releases surrounding the control building, (4) protection against the effects of high-energy line ruptures in adjacent plant areas, and (5) fire protection to ensure that the control room is manned continuously. In CESSAR-DC Chapter 15, ABB-CE describes the methods to limit the amount of radiation that control room personnel will be exposed to under accident conditions. Similarly, in CESSAR-DC Section 9.5.1, ABB-CE describes fire protection methodology, which the staff evaluated as documented in Section 9.5.1 of this report.

The control room emergency zone (CREZ) consists of the control room, the reactor operator's office, the control room supervisor's office, the emergency supplies room, the integrated plant status overview room, and the document room. In CESSAR-DC Table 3.2.1, ABB-CE states that the vital instrumentation and equipment rooms (including battery rooms) and the components of the air conditioning system for the MCR (including the air handling units and its filters, fans, ductwork, water-cooling coils, and heating coils) are Safety Class 3 and seismic Category I; will meet the quality assurance requirements of Appendix B to 10 CFR Part 50; and will remain functional after a safe-shutdown earthquake. Intake and exhaust structures will be protected from tornado-generated missiles, wind-generated missiles, rain, snow, or trash.

The MCR air conditioning system is a safety related system consisting of an air conditioning system and emergency filtration system. The system has sufficient redundancy to ensure operation under emergency conditions, assuming the single failure of any one component. Redundant safety related components of the system are physically separated and protected from internally generated missiles, pipe breaks, and water sprays. The facility has backup power sources for the safety-related components of the systems. The control room habitability systems descriptions, design parameters, and figures are provided in CESSAR-DC Sections 6.4 and 9.4-1; Ta-

bles 9.4-1, 9.4-3, 9.4-3A, 9.4-5 and 15A-10; and Figures 9.4-1 and 9.4-2, respectively. In Section 9.4-1, the staff evaluates the control complex ventilation system. In CESSAR-DC Table 9.4-3A, ABB-CE details its conformance to the guidance of SRP Table 6.5-1 concerning the minimum instrumentations and controls. In CESSAR-DC Table 9.4-5, ABB-CE delineates its conformance with the guidance of RG 1.52.

ABB-CE responded to the staff's RAI Q450.3, concerning the outside air supply for pressurization during emergency mode recirculation through the filter unit, including specifying the locations of the control room inlets, the exhaust vents, and the plant vents; describing the CREZ envelope; and stating the rate of system unfiltered in-leakage infiltration. ABB-CE also responded to the staff's RAI Q410.116, concerning the air intake isolation dampers; air filtration unit bypass dampers; redundant radiation and toxic gas monitors; smoke detectors; fresh air intakes protection against the effects of high winds, rain, snow, ice, and trash; duct testing; hydrogen concentration inside battery rooms; and pre-and post-HEPA filters. The information is acceptable as discussed below.

ABB-CE made a commitment to prepare revised information and incorporate it into the CESSAR-DC as follows:

- (1) Revisions to general arrangement drawings (CESSAR-DC Figures 1.2-3 and 1.2-8) to incorporate the physical locations of the control room inlet and exhaust vents on the nuclear annex building in relation to the containment structure. This was identified as DSER Confirmatory Item 6.4-1.
- (2) Revisions to general arrangement drawings (CESSAR-DC Figures 1.2-3 and 1.2-11) to incorporate the location of the unit vent. This was identified as DSER Confirmatory Item 6.4-2.
- (3) An amendment to the CESSAR-DC to incorporate ABB-CE's responses to RAI Q450-3 (Section 6.4) and Q410.116 (Section 9.4.1). This was identified as DSER Confirmatory Item 6.4-3.

By CESSAR-DC amendments, ABB-CE stated that the locations of the control room air intakes and plant unit vent are shown in the general arrangement drawings, CESSAR-DC Figures 1.2-3 and 1.2-8, and 1.2-3 and 1.2-11 respectively. Therefore, the above Confirmatory Items 6.4-1 and 6.4-2 are resolved.

Subsequently, ABB-CE incorporated responses to RAIs 450.3 and 410.116 concerning the issues in DSER Confirmatory Item 6.4-3 as follows:

- (1) Makeup air of 0.94 m³/sec (2000 cubic ft per minute (cfm)) is provided from the least contaminated control room air intake to offset the maximum anticipated outleakage of 0.94 m³/sec (2000 cfm) and to pressurize the control room to a minimum of 3.2 mm (1/8-in.) water gauge positive pressure with respect to the adjacent areas.
- (2) The plant unit vent and diesel building exhausts are located at least 61 m (200 ft) away from the nearest control room intake (as shown in CESSAR-DC Figure 3.8-5).
- (3) The CREZ volume is 1906 m³ (67,300 ft³) and the maximum unfiltered infiltration rate into the CREZ under accident conditions is 0.005 m³/sec (10 cfm) (as shown in CESSAR-DC Table 15A-10). CREZ consists of the control room, reactor operator's office, control room supervisor's office, emergency supplies room, integrated plant status overview room, and document room.
- (4) The control complex ventilation system has two motor-operated isolation dampers for the outside air intakes, two pneumatically operated (bypass) dampers designed to fail closed, and redundant radiation and toxic gas monitors in each division for the filtration and toxic function. Both motor operated dampers will not be simultaneously closed on the detection of radioactive materials at both air intakes. One air intake will remain open to provide (pressurization) make-up air to balance the exfiltration. Air intakes are protected against the effects of tornado and wind-generated missiles, rain, snow, ice or trash. The divisional filtration unit consists of a post-filter, which is a HEPA, downstream of the carbon adsorber (as shown in CESSAR-DC Figure 9.4-1).
- (5) The system ductwork is leak tested in accordance with ASME N509. The battery room exhausts maintain the hydrogen concentration in the battery room below two percent.

On this basis, DSER Confirmatory Item 6.4-3 is resolved.

The air intakes are located on opposite sides of the building but are not separated by 180 degrees. Each outside air inlet has two isolation dampers, redundant toxic gas and radiation monitors, and a smoke detector. The emergency filtration system starts automatically if high radiation is detected at an air intake, or if a SIAS is received. If high radiation is detected at both air intakes, the automatic selection logic compares the radiation levels at each air intake and closes the isolation dampers in the air intake

which has the higher reading. Therefore, outside air to pressurize the control room comes through the comparatively less contaminated inlet automatically. The pressurization and recirculation modes can also be actuated manually from the control room.

During an accident, the system operates in a pressurized mode, drawing in 0.94 m³/sec (2,000 cfm) of outside air which is mixed with 1.89 m³/sec (4000 cfm) of air recirculated from the control room prior to being filtered by the control room filter unit. The emergency zone volume is 1,906 m³ (67,300 ft³). The entire flow rate of 2.83 m³/sec (6,000 cfm) passes through the filter unit which includes a moisture separator, prefilter, electric preheater, absolute filter (HEPA), carbon adsorber [activated carbon depth of the carbon filter is 51 mm (2 in.)], postfilter (HEPA), ducts and valves, and a fan as shown in CESSAR-DC Figures 9.4-1 and 9.4-2. The charcoal tray and screen will be all welded construction to preclude the potential loss of charcoal from adsorber cells in accordance with IE Bulletin 80-03. All ducts and equipment housings outside the CREZ are of welded construction. Flanged connections will be pressure tight and periodically visually examined and tested to maintain a positive pressure with respect to the adjacent areas so that any unfiltered inleakages inside CREZ are precluded. The system is designed to maintain an infiltration rate during pressurized operation of less than 0.005 m³/sec (10 cfm). No steam piping adjacent to CREZ air intakes or inside CREZ exists and no other HVAC system ducts other than MCR air-handling system ducts pass through the CREZ.

In CESSAR-DC, ABB-CE states that the habitability systems will maintain a continuous pressure in the control room envelope at a value that is positive, and slightly greater than the pressure outdoors and in surrounding areas. The staff does not agree with ABB-CE concerning the proposed pressurization of the CREZ. The design does not conform to the acceptance criterion in SRP Section 6.4 regarding the pressurization systems. The SRP requires verification that the control room boundary is pressurized to a value of at least 3.2 mm (1/8-in.) water gauge greater than the pressure in all surrounding air spaces while applying makeup air at the design rate. This was identified as DSER Open Item 6.4-1.

Subsequently, ABB-CE stated that the control room boundary pressurization system will be periodically tested (every 18 months) to verify that the makeup up air required to maintain a minimum of 3.2 mm (1/8-in.) water gauge pressure inside the control room boundary with respect to the adjacent areas does not exceed 10 percent of the design value. On this basis, DSER Open Item 6.4-1 is resolved.

Engineered Safety Features

The staff assessed the capability of the control room filter unit to limit the exposure of the control room operator in the event of an accident utilizing the source terms presented in Chapter 15 of this FSER. In determining the impact of radioiodine on the control room operators, the staff calculated an iodine protection factor (IPF) in accordance with SRP Section 6.4. Based upon 0.472 m³/sec (10 cfm) of unfiltered inleakage, relative humidity control of the filtered air, a 5 cm (2 in.) bed of charcoal, a HEPA filter, 0.944 m³/sec (2000 cfm) of filtered outside makeup air and 1.888 m³/sec (4,000 cfm) of filtered recirculation flow, the IPFs were approximately 53 for elemental and organic forms of radioiodine and approximately 200 for the particulate form of radioiodine. The staff determined that, because of the dual inlets for the control room intakes and the automatic selection of the intake with the lower radioactivity level, the control room atmospheric dilution factor (X/Q) value could be reduced by an additional factor. However, because the intakes were not separated by 180° as recommended by SRP Section 6.4, the X/Q reduction factor of 10 could not be applied for the short time periods associated with an accident. It was only after a 4 day waiting period that the factor of 10 could be applied.

The staff did not independently calculate the control room operator doses; however, the staff performed an independent assessment of the atmospheric relative concentrations (χ /Qs) for the control room habitability assessment.

In Section 2.3.6 of CESSAR-DC, ABB-CE calculated the on-site accident one hour χ /Q values using a time-based building wake mode. This model was developed by Ramsdell and was described in Atmospheric Environment, Volume 24B, No. 3, pp 377-388, 1990, (Ramsdell, J.V., Diffusion in Building Wakes for Ground-Level Releases). Solely for the purposes of determining the χ /Q value associated with the limiting accident scenario, ABB-CE used a meteorological database that was characterized as "representative of an 80-90th percentile of U.S. commercial nuclear power plant sites." The resulting χ /Q values were provided in CESSAR-DC Tables 2.3-2 through 2.3-5.

In conjunction with the χ /Q values, ABB-CE used dose conversion factors based on International Commission on Radiological Protection (ICRP) Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation." The postulated control room doses using the 1990 Ramsdell/ICRP 2 framework were provided in CESSAR-DC Table 6.4-1 and compared to the dose guidelines in SRP 6.4. Source terms for a suite of design basis accident scenarios were determined and the χ /Q values were scaled upward to the value necessary to set the postulated dose at the SRP 6.4 limiting dose guideline value.

In May 1994, PNL conducted a peer review meeting to assess the Ramsdell Model. At that meeting the peer group expressed concerns regarding the model's treatment of receptors within three building heights of the release point. The peer review group identified information for close-in distances that could be utilized to evaluate the performance of the Ramsdell Model. This work is the near-source dilution model developed by Wilson and Chui. (Wilson, D.J. and E.H. Chui, Influence of Building Size on Rooftop Dispersion of Exhaust Gas, Atmospheric Environment, Volume 28, 1994, accepted for publication).

One objective of the Wilson-Chui Model is to predict a lower-bound minimum dilution function (maximum χ /Qs). The field measurements form the experimental data used to derive the 1990 Ramsdell Model were used to evaluate the performance of the Wilson-Chui Model as a predictive tool. The results of this comparison showed that use of the Wilson-Chui Model resulted in χ /Q values which were greater than the experimental data in every case. For the purposes of evaluating ABB/CE's use of the 1990 Ramsdell Model, the staff considered the Wilson-Chui Model results at a performance level whereby it would not be exceeded by more than 5 percent of the experimental data.

In conjunction with the 95th percentile Wilson-Chui Model χ /Q values, the staff used dose conversion factors based on ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers." The staff determined that, using the Wilson-Chui/ICRP 30 framework, the control room habitability aspects of the ABB-CE System 80+ design are acceptable. For the ABB-CE System 80+ design, as long as the linkage exists between the use of the 1990 Ramsdell Model and ICRP 2, the staff concludes that control room habitability is acceptable.

ABB-CE has submitted a set of design description and ITAAC relating to the control complex ventilation system. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. In CESSAR-DC Table 3.2-4, ABB-CE states that the control areas structure will be protected against the environment including: soil and water pressure on the substructure; tornado pressure drop; thermal stresses; and pipe and pipe rupture loads. In CESSAR-DC ABB-CE states that no toxic gases, steam, or radioactive materials will be piped or stored near the CREZ. ABB-CE also states that the flood protection measures for seismic Category I SSCs are designed in accordance with RG 1.102, "Flood Protection for Nuclear Power Plants."

Additionally, ABB-CE states that (1) internally generated missiles (outside the containment) from rotating and pressurized components are not considered credible as

described in CESSAR-DC Section 3.5.1.1, and (2) no high-energy lines will be near the control room. Therefore, ABB-CE concludes that the habitability systems will be protected against dynamic effects that may result from possible failures of such lines. In Sections 3.4.1, 3.5.1.1, 3.5.2, and 3.6.1 of this report, the staff documents its evaluation of the protection against floods, internally and externally generated missiles, and high- and moderate-energy pipe breaks. The staff concludes that the control room habitability systems comply with GDC 4 "Environmental and Dynamic Effects Design Bases," as it relates to protection of the system against floods, internally generated missiles and piping failures.

The provision for seismic Category 1, Class 1E chemical instrumentation to detect toxic releases at air intakes if needed, is included in CESSAR-DC Section 6.4.2.2. In accordance with TMI Action Plan Item III.D.3.4, a COL applicant referencing the System 80+ design must demonstrate that control room operators are adequately protected against the effects of the release of toxic substances, either on or off the site; and that the plant can be safely operated or shut down under conditions created by any design-basis accident. The COL applicant must determine the amounts and locations of any possible sources of toxic substances near the plant using the methods in RGs 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." The COL applicant must include specific detectors where necessary to permit automatic isolation of the control room. The protection against toxic gases to ensure that control room operators are protected against releases of hazardous material was identified as a COL Action Item 6.4-2 in the DSER. Subsequently, in CESSAR-DC, ABB-CE states that the COL applicant shall verify that the following site specific requirements are met to ensure adequacy with the System 80+ standard design (1) perform the toxic chemical release analysis; (2) prepare detailed operating procedures to cope with the toxic gas accident; and (3) provide the location and detailed design of the breathing air storage reservoir. The COL applicant will also develop operating procedures to cope with the closure of both air intakes due to a high concentration of smoke. On this basis, COL Action Item 6.4-2 is acceptable.

The COL applicant will need to verify that the as-built design; the operating, maintenance, and emergency procedures and training; and the performance characteristics of the control room habitability system are consistent with the licensing basis documentation. The COL applicant will also need to verify that the TS and surveillance

procedures are consistent with the licensing basis documentation. This verification was identified as a COL Action Item 6.4-3 in the DSER. Subsequently, in CESSAR-DC, ABB-CE states that the COL applicant will verify that the control room habitability system is consistent with the licensing basis documentation for the above requirements. On this basis, COL Action Item 6.4-3 is acceptable.

Based on the above, the staff concludes that the control room habitability systems meet the acceptance criteria of SRP Section 6.4 and are, therefore, acceptable.

6.5 Containment Spray System

The containment spray system (CSS) is a safety-grade system designed to reduce containment pressure and temperature following a LOCA or MSLB and to remove fission products from the containment atmosphere during a LOCA. The staff reviewed the design basis for the fission product removal function of the CSS to verify that it is consistent with the assumptions made in the accident evaluations of CESSAR-DC Chapter 15. The acceptance for the fission product cleanup function of the containment spray system is based on conforming to the relevant requirements and criteria in SRP Section 6.5.2 as it relates to determining the fission product removal effectiveness whenever the applicant claims a containment atmosphere fission product cleanup function for the systems, and GDC 41 as it relates to containment atmosphere cleanup systems being designed to control fission product release to the reactor containment flowing postulated accidents, GDC 42 as it relates to containment atmosphere cleanup systems being designed to permit appropriate periodic inspections, and GDC 43 as it relates to containment atmosphere cleanup systems being designed for appropriate periodic functional testing.

The CSS consists of two 100-percent-capacity redundant and separated trains, each of which has one pump, one heat exchanger, and associated spray nozzles, spray headers, piping, valves, and instrumentation and controls. The CSS pumps start upon receiving a SIAS or containment spray actuation signal and the spray header isolation valves open upon receiving a containment spray actuation signal. The CSS pumps are designed to be functionally interchangeable with the SCS pumps to provide residual heat removal or to provide cooling of the IRWST. This will allow each pump to provide a back-up when the pumps are not needed for their requisite function. The CSS pumps and heat exchangers can be manually aligned to provide LTC of the IRWST during post-LOCA feed-and-bleed operations when the steam generators are not available to cool the RCS.

Engineered Safety Features

During a LOCA, the CSS draws borated water from the IRWST and sprays to the containment atmosphere from the upper region of the containment. The sprayed water will ultimately drain back to the IRWST through the HVT. The HVT is a 227,000 L (60,000 gal) capacity stainless steel-lined tank next to the IRWST with drainage spillway connections to the IRWST. Both the IRWST and HVT are reinforced concrete structures with stainless steel liners on the surfaces in contact with borated water. The sprayed water returns to the IRWST through the spillways once the HVT water level reaches the spillway. The IRWST is a continuous water source during short-term injection and long-term cooling modes of postaccident operation. There is no recirculation spray mode in this design. The effectiveness of the CSS in removing iodine from the containment is addressed in Chapter 15 of this report.

The borated spray water contains no additive for pH control during the initial stage of a LOCA. After the blowdown, coolant from the LOCA accumulates in the HVT and starts to flow into the IRWST causing the water in the IRWST to become less acidic. The water in the IRWST is brought to a minimum pH of 7 for post-LOCA iodine retention. Postaccident pH control of the spray water in the IRWST initially was to be provided by the granular disodium phosphate base compound which is stored in stainless steel baskets in the HVT. In Amendment R, ABB-CE changed the spray additive chemical from disodium phosphate to trisodium phosphate dodecahydrate and revised CESSAR-DC Sections 6.5.1.1, 6.5.1.2, 6.5.3.3, 16.8.5, and 16A.8.5, and Figures 6.5.4, 6.5.5, and 6.8.2, to reflect the change of spray additive. The volume of trisodium phosphate dodecahydrate required to establish a minimum pH of 7 in the recirculated containment spray solution was calculated to be 926 cubic feet based on a maximum allowable boron concentration of 4400 ppm boron. The staff finds that trisodium phosphate dodecahydrate is a stronger base alkaline than disodium phosphate. The change will enhance the capability of maintaining the IRWST water at a pH above 7 and, therefore, is acceptable.

The elevation of the baskets is above the normal operating water level in the HVT and below the IRWST spillways. During a LOCA, the baskets become immersed in water and the resulting solution overflows into the IRWST. Therefore, the spray fluid will become less acidic as the trisodium phosphate dodecahydrate mixes with the boric acid.

SRP Section 6.5.2, Item II.1.g, states in part, that long-term iodine retention may be assumed only when the equilibrium sump solution pH, after mixing and dilution with the primary coolant and ECCS injection, is above 7. In the SRP, the staff states that this pH value should be

achieved by the onset of the spray recirculation mode. In response to the staff's concern regarding the long-term pH of the spray solution after a LOCA, ABB-CE stated that the pH will be brought to 7, as discussed above. In addition, the containment and the CSS are designed to withstand the chemical environment imposed by spraying borated water from the IRWST and any subsequent long-term induced chemical environments. Therefore, the pH of the spray solution conforms to the SRP requirement.

Each of the two independent CSS trains has its own spray header and nozzles located in the upper part of the containment. The spray nozzles (SPRACO Company Model 1713A) are a non-clogging type and can pass particles up to 8 mm (5/16 in.) in diameter while covering 90 percent of the containment area with a maximum drop fall height of 25 m (83 ft) and an average drop residence time of 13 seconds. The design mean drop size is 530 microns and the median drop size is 230 microns. In response to a staff question, ABB-CE submitted a histogram confirming the size distribution of the spray nozzle droplets as measured under nozzle design conditions by the manufacturer. However, ABB-CE did not specify the location of the spray nozzles and, therefore, the staff could not verify that 90 percent of the containment free volume would be covered by the sprays. This was designated as DSER Open Item 6.5-1.

In Amendment N, ABB-CE revised the CESSAR-DC to state that the containment spray headers and nozzles have been located inside the containment so that 65 percent of the containment free volume is sprayed rather than the previously stated value of 90 percent. The spray removal constant for elemental iodine was re-evaluated in accordance with SRP Section 6.5.2 (Rev. 2) based on the new 65 percent volume spray coverage to yield a new value of 20 lambda per hour. ABB-CE also made major changes in CESSAR-DC Section 6.5 by adding additional subsections and revised tables and figures in conformity with the reanalyses. Subsequently, ABB-CE changed the elemental iodine spray removal constant to 10 to 13 lambda per hour within 30 to 110 minutes.

The revised CESSAR-DC contains all the pertinent information regarding the modified header and nozzle arrangement, as well as the new spray removal constant for elemental iodine. The spray volume is divided into three regions, Regions I, II, and III, which represent 82 percent of the total containment free volume. The free volumes not included as part of these regions are combined together in Region IV as unsprayed volume, which is about 18 percent of the total containment free volume. ABB-CE states that since about 80 percent of the free volume in each of the regions is actually covered by spray drops, the effective spray volume of the three sprayed regions is

estimated at 65 percent of total containment free volume. Consequently, the remaining 35 percent of containment free volume is assumed unsprayed. Mixing between the sprayed and unsprayed volume of the containment was calculated with a minimum mixing rate as shown in CESSAR Figure 6.5.4. ABB-CE states that the spray nozzles and headers are oriented to provide maximum effective coverage of the containment volume.

However, ABB-CE reevaluated the offsite and control room LOCA dose analyses in CESSAR-DC Chapter 15 using the 82 percent spray coverage and revised the elemental iodine removal constant in CESSAR-DC Figure 6.5-5. The staff found that the spray volume used in the CESSAR-DC Chapter 15 calculation was inconsistent with the effective spray volume of 65 percent stated in CESSAR-DC Section 6.5.

In a conference call, ABB-CE stated that the 82 percent spray volume is the effective two-control volume containment model value used for elemental iodine removal constant calculation in the offsite dose and control room LOCA dose analyses. The staff asked ABB-CE to reevaluate the containment spray capability based on 82 percent spray coverage for all cases. Subsequently, ABB-CE changed the effective spray volume in CESSAR-DC Section 6.5 from 65 percent to 82 percent to be consistent with the Chapter 15 analysis. On this basis, DSER Open Item 6.5-1 is resolved.

The minimum NPSH of the CSS pumps was calculated to be 6.5 m (21.3 ft) at pump inlet flow during all phases of an accident based on a low elevation and location of the pumps relative to the IRWST. In calculating the available NPSH, ABB-CE assumes both safety injection and containment spray pumps will operate concurrently and does not take credit for water that could be trapped inside the containment and not available to the IRWST. In the DSER, the staff stated that there was no comparison between the calculated minimum CSS pump NPSH of 6.5 m (21.3 ft) and a design NPSH requirement since no specific CSS pump has been selected for this design. This was designated as DSER Open Item 6.5-2.

ABB-CE states that the NPSH design requirement has been addressed in the revised CESSAR-DC Section 6.5.3.4 (Amendment N). In CESSAR-DC Section 6.5.3.4, ABB-CE states, in part, that the IRWST is the suction source for both SI and CSS pumps during short-term injection and long-term cooling modes of postaccident operation. The minimum available NPSH for the SI and CSS pumps was determined on based of the minimum water level in the IRWST during accident conditions. The SI and CSS pumps are located in the reactor building subsphere and

are placed low enough below the minimum IRWST water level to ensure adequate available NPSH. The minimum IRWST water level elevation was determined to be 23 m (75.5 ft). The calculated available NPSH for the CSS pumps ranges from 7.3 m (24 ft) at the design flow rate of 18,900 L/min (5000 gpm) to 6.5 m (21.2 ft) at a pump runout flow of 24,600 L/min (6,500 gpm). This exceeds the CSS pump required NPSH of 6.1 m (20 ft) at runout flow. In addition, the reactor cavity will not be flooded during a LOCA as was assumed in determining the minimum water level of 23 m (75.5 ft) in the IRWST. Because of this, an additional volume of water will raise the minimum water level in the IRWST by approximately 0.6 m (2 ft), thereby increasing the available NPSH. ABB-CE also addressed the prevention of debris blocking the return water from entering of the IRWST in CESSAR-DC Section 6.8. On the basis of this review, the staff finds that there will be enough water in the IRWST to cover the minimum CSS pump NPSH. Therefore, DSER Open Item 6.5-2 is resolved.

In CESSAR-DC Section 6.5.4, ABB-CE states that the COL applicant will conduct preoperational tests on the CSS to verify the operability of all pumps, valves, spray headers, spray nozzles, heat exchangers, and instrumentation. Before operating the plant, the COL applicant will also conduct a series of hydrostatic tests in accordance with the ASME Boiler and Pressure Vessel Code; this was designated as COL Action Item 6.5-1 in the DSER and is acceptable.

The CSS is designed to withstand the impact of floods, pipe breaks, missiles, pipe whip, water hammer, overpressurization, fire, and a LOOP. Each CSS train has a separate power supply through an independent electrical division and emergency diesel generator (EDG). Physical separation is used throughout the system to preclude common-mode failures and enhance fire protection capabilities. CSS pipes are made of austenitic stainless steel, conform to ASME Boiler and Pressure Vessel Code, Section III, and are classified as seismic Category I piping system. All components of the CSS will be accessible for maintenance, inspection, testing, and manual operation even after a design-basis LOCA. All CSS-related equipment not covered by ASME Code Section III will also be accessible.

To confirm the effectiveness of the CSS, ABB-CE performed CSS failure modes and effects analysis (FMEA) and presented the results in CESSAR-DC Table 6.5-3. The analysis identified 14 different component failures. In most cases, the compensating feature of a parallel redundant containment spray path or train ensures that the spray function will not be degraded by any failure. The remaining cases have no effect on the operability of the CSS.

Engineered Safety Features

In CESSAR-DC Appendix 15A, ABB-CE addresses the performance of the CSS in removing fission products and presented the method it used to calculate the radiological consequences of accidents. ABB-CE recalculated the CSS fission product removal capability during a design-basis LOCA based on the following input to the analysis:

Fraction of net free containment volume being sprayed = 82 percent

Transfer rate between sprayed and unsprayed regions = volume of unsprayed region per hour per CESSAR-DC Fig. 6.5-4

Elemental iodine spray removal constant = 10-13 lambda/hour within 30-110 minutes, reduced by a factor of 1.26 to account for mixing with the sprayed region. (see CESSAR-DC Fig. 6.5-5)

The staff reviewed this analysis in accordance with SRP Section 6.5.2 and its referenced standard ANSI/ANS-56.5-1979 and confirmed the use of an elemental iodine spray removal coefficient of 13 lambda per hour which is within the limit specified in SRP Section 6.5.2. Heat removal by sprays does not become dominant until after the first 10 minutes. Therefore, containment mixing is limited to two unsprayed volumes per hour during the first 10 minutes. The staff finds that the use of minimum mixing rate during the immediate post-blowdown period is conservative.

In the CESSAR-DC, ABB-CE credits only the CSS for removing the elemental form of iodine from the containment atmosphere after a design-basis LOCA. Following a design-basis LOCA, any leakage from the containment atmosphere is held up and processed in the annulus and the reactor building subsphere by the AVS and subsphere ventilation system (SVS) respectively. The AVS and SVS are ESF atmosphere cleanup systems and are addressed in Section 6.2.3 and Section 9.4.5 of this report.

In CESSAR-DC Section 15.6.5, ABB-CE presents the offsite dose results of the design-basis LOCA analysis due to containment leakage and annulus ventilation discharge. The calculated 2-hour thyroid and whole-body doses for the exclusion area radius are 1.83 Sv (183 rem) and 26.7 mSv (2.67 rem), respectively. These doses are less than the limits in 10 CFR Part 100 of 3 Sv (300 rem) thyroid and 250 mSv (25 rem) whole body and are, therefore, acceptable.

In the DSER, the staff stated that ABB-CE had not estimated the mean resident time of soluble volatile and particulate fission products in the containment building atmosphere after an accident, and had not indicated how long the continuous spray will last. Acceptance Criterion II.1.a in SRP Section 6.5.2 states, in part, that the operating period of the containment spray system should not be less than 2 hours in all cases. The staff stated that ABB-CE is required to verify the CSS postaccident operation period, which should be no less than 2 hours in all cases. This was designated as DSER Open Item 6.5-3.

In CESSAR-DC ABB-CE states that the containment spray system is designed to operate throughout the duration of a DBA, as indicated in CESSAR-DC Table 3.11B-1, up to 100 days without interruption. Since the operating period of the spray is far more than two hours, DSER Open Item 6.5-3 is resolved.

In Amendment W, ABB-CE revised the CESSAR to add Section 6.5.5. The new section addresses the emergency containment spray backup system (ECSBS), which is used to provide containment spray in the event of a beyond design basis accident in which both CS pumps and SCS pumps, and/or the IRWST are assumed not available. The ECSBS provides a T-connection for a temporary water source outside the Nuclear Annex to supply containment spray after a severe accident to prevent a catastrophic failure of the containment. The staff's evaluation of the ECSBS is provided in Section 19.2.3.3.2.1.3 of this report.

On the basis of its review, the staff concludes that ABB-CE System 80+ containment spray system, as a fission product cleanup system, is acceptable and complies with the requirements of (1) GDC 41 with respect to the iodine removal function following a postulated loss-of-coolant accident; (2) GDC 42 with respect to the capability for periodic inspection of the system, and (3) GDC 43 with respect to the capability for periodic testing of the system.

6.6 Inservice Inspection of Class 2 and 3 Components

The staff has reviewed the ASME Code Class 2 and 3 ISI requirements stated in the CESSAR-DC to the criteria in SRP 6.6. The areas reviewed included the components subject to examination, accessibility, examination categories and methods, inspection intervals, evaluation of examination results, system pressure tests, augmented ISI to protect against postulated piping failures, code exemptions, and relief requests. IST is evaluated separately in Section 3.9.6 of this report. Open items in the DSER are addressed as follows:

ABB-CE, in Amendment N of the CESSAR-DC, required that Class 2 and 3 components requiring inspection will be designed and provided with access to enable the performance of Section XI inspections on site. On this basis, DSER Open Item 6.6-1 is resolved.

ABB-CE, in Amendment N, revised the CESSAR-DC to require that preservice inspection (PSI) will comply with Section XI of the same edition as the ASME Code used for construction and ISI will meet Section XI in effect in accordance with 10 CFR 50.55a(g). On this basis, DSER Open Item 6.6-2 is resolved.

In response to RAI Q252.17, ABB-CE indicates that relief requests will be submitted to the NRC when Section XI PSI requirements are impractical. However, because the PSI requirements are established and known at the time each component is ordered, 10 CFR 50.55a(g) does not have provisions for relief requests for impractical PSI examination requirements. ABB-CE in Amendment N to the CESSAR-DC specified that all PSI requirements will be in accordance with Section XI, of the same edition as the ASME Code used for construction. On this basis, DSER Open Item 6.6-3 resolved.

The COL applicant has the overall responsibility for performing the PSI and ISI of Class 2 and 3 components throughout the service life of the plant. However, the plant should be designed for accomplishment of pre-service and ISI, and in particular, should reflect the requirements of Appendix VIII to Section XI of the ASME Code, which ABB-CE has acknowledged.

ABB-CE modified Section 6.6.2 of CESSAR-DC to state that systems and components are designed so that design, materials, and geometry do not restrict inspections required by Section XI of the ASME Code. The basis for requests for relief is in 10 CFR 50.55a(g)(4). ABB-CE has eliminated the need for relief under this paragraph, and accordingly, requests for relief by COL applicants because materials, design, or geometry make ISI examinations and tests impractical should not occur. A review of Palo Verde, Units 1, 2, and 3, history of relief requests in "Commercial Nuclear Power Plant ISI Relief Request Data Base Regarding ASME Code Section XI Requirements," prepared by The Idaho National Engineering Laboratory, showed that no requests for relief have been granted to date for material or design limitations. Two were granted for reason of limited access (removal of insulation) which, under the 1989 edition of the Code, would not be an acceptable reason for granting relief in that this particular edition of the Code requires access be provided to accomplish PSI and ISI examinations. The staff views the Palo Verde experience as evidence that ABB-CE has maintained a philosophy of designing its units so that code-required

inspections can be accomplished. On this basis, DSER Open Item 6.6-4 is resolved.

ASME Section XI indicates that PSI should be conducted with equipment and techniques equivalent to those that are expected to be used for subsequent ISI. The PSI provides the baseline information for reference in the subsequent ISI. For example, if the ISI of a piping weld is expected to be performed by ultrasonic techniques, the PSI should also be based on ultrasonic techniques. ABB-CE in Amendment N to the CESSAR-DC, stated its commitment to this requirement. On this basis, DSER Open Item 6.6-5 is resolved.

The ASME has published in Section XI (Division I), Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The NRC has approved the 1989 Edition of the ASME Code Section XI which includes Appendix VII. The applicable Code Edition for CESSAR-DC is the 1989 Edition which incorporates Appendix VII. ABB-CE revised the CESSAR-DC with Amendment N to require that the Section XI requirements are to be augmented with the requirements in Appendix VIII. On this basis, DSER Open Item 6.6-6 is resolved.

ASME Class 1, 2, and 3 carbon and low-alloy-steel piping items that are susceptible to wall thinning due to the single-phase (water) erosion-corrosion phenomenon will be subject to examination in accordance with Subsection IWH of ASME Section XI. Subsection IWH of ASME Section XI has not been incorporated in the code at the time of this review. However, ABB-CE in a submittal dated December 18, 1992, committed to review Subsection IWH and ensure that the System 80+ design PSI and ISI programs comply accordingly. Where the design will not practically accommodate full adherence to IWH, alternatives will be proposed in accordance with 10 CFR 50.55a(a)(3). On this basis, DSER Open Item 6.6-7 is resolved.

To ensure that no deleterious defects develop during service in ASME Code Class 2 components, selected welds and weld heat-affected-zones are inspected before reactor startup and periodically throughout the life of the plant. In addition, Code Class 2 and 3 systems receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

In CESSAR-DC, ABB-CE states that its inspection program will comply with the rules published in 10 CFR 50.55a and Section XI of the ASME Code. The ISI program will consist of a preservice inspection plan and an ISI plan.

Engineered Safety Features

Examples of Code Class 2 systems are residual heat removal systems, portions of chemical and volume control systems, and ESFs not part of the Code Class 1 systems. Examples of Code Class 3 systems are component cooling water systems and portions of radwaste systems. All of these systems transport fluids.

The staff concludes that the ISI program is acceptable and meets the inspection and pressure testing requirements of GDC 36, 37, 39, 40, 43, 45, and 46 and 10 CFR 50.55a.

The COL applicant should submit PSI and ISI program plans for staff review and approval. The COL applicant should verify that its PSI and ISI programs will incorporate the requirements of Appendices VII and VIII and Subsection IWH of ASME Code Section XI. This was identified as COL Action Item 6.6-1 in the DSER and is acceptable.

6.7 Safety Depressurization System

In CESSAR-DC Section 6.7, ABB-CE describes the safety depressurization system (SDS), which consists of the reactor coolant gas vent system (RCGVS) and the RDS. The SDS performs two functions: venting the RCS and rapidly depressurizing the RCS.

6.7.1 Reactor Coolant Gas Vent System

The RCGVS is a manually operated safety-grade system. It removes noncondensable gases or steam from the pressurizer through a vent line to the reactor drain tank (RDT) through one or both of the parallel isolation valves. The RCGVS can also remove noncondensable gases or steam from the reactor vessel upper head through a vent line to the RDT through a flow-restricting orifice and one or both of the parallel isolation valves. The RCGVS isolation valves are closed during normal reactor operation. During shutdown or transient conditions, the operator will follow operating procedures to manually actuate (open) the RCGVS valves from the MCR in order to vent the reactor vessel upper head or the pressurizer steam space if the operator determines that noncondensable gases have collected in the reactor vessel upper head or in the pressurizer steam space. The RCGVS will have the capability to be manually actuated, monitored, and controlled from the control room, as required by GDC 19.

The parallel isolation valves are powered by a normal ac power source and an emergency ac power source. In CESSAR-DC Table 6.7-3, ABB-CE documents a failure modes and effects analysis (FMEA) demonstrating that the RCGVS will maintain a vent path after a single failure of either of the valves or of the power source. This demonstration complies with the requirements of GDC 17 and 34.

To satisfy the requirements for natural circulation cooldown specified in SRP, BTP RSB 5-1, the RCGVS can remove the steam bubble and depressurize the RCS for the shutdown cooling mode of operation. This is discussed further in Section 5.4.3-1 of this report.

The RCGVS is designed to seismic Category I requirements in compliance with RG 1.29.

The RCGVS is designed to permit periodic inspection in accordance with

Section XI of the ASME Boiler and Pressure Vessel Codes. This constitutes compliance with GDC 36.

In response to the staff's request (RAI Q440.134), ABB-CE committed to include the RCGVS as a safety-grade system in the System 80+ TS. This commitment is acceptable. This was designated as DSER Confirmatory Item 6.7.1-1.

In CESSAR-DC Section 16.7.17 (Amendment O), ABB-CE includes a TS with the limiting condition for operation and surveillance requirements (SRs) for the RCGVS. The SRs include verification of the flow path, cycling the vent valves from the control room, and verification of correct breaker alignment and power sources for valve position indications. The staff reviewed the TS for RCGVS and found that the TS provides reasonable assurance for the operability of RCGVS. The staff concludes that the TS is acceptable. On this basis, DSER Confirmatory Item 6.7.1-1 (regarding the TS for RCGVS) is resolved.

The staff requested that ABB-CE submit emergency procedure guidelines (EPGs) on the RCGVS. This was designated as DSER Open Item 6.7.1-1. In response, ABB-CE included RCGVS in the EOGs for removal of the noncondensable gases from the reactor vessel head. This instruction is implemented in step 37 of the excess steam demand event recovery guideline, step 28 of loss of the all feedwater recovery guideline, step 32 of the steam generator tube rupture recovery guideline, step 30 of the loss of offsite power recovery guideline, steps 20 and 55 of the LOCA recovery guideline (LOCARG), and step 37 of HR-3, step 3 of HR-5 in the FRG. ABB-CE also added instructions in steps 21 and 42 of the LOCARG, step 17 of HR-3 in the FRG for depressurization control by using the RCGVS. The staff finds the EOG steps adequate for using the RCGVS for event mitigation. On the basis, DSER Open Item 6.7.1-1 is resolved.

DSER Open Item 6.7.1-2 required ABB-CE to confirm the required flow capacity of the RCGVS. In response, ABB-CE included a test in the ITAAC for the SDS to determine the RCS depressurization rate using the pressurizer portion of the RCGVS function. The test will be performed at

normal operating pressure of 1.55×10^4 kPa (2250 psia) to confirm that the depressurization rate is not less than 6.2 kPa (0.9 psia/sec), which is the rate with the maximum flow rate passing through the RCGVS valve calculated in the natural circulation cooldown analysis in CESSAR-DC Appendix 5D. An adequate test is included in the SDS ITAAC and CESSAR-DC Section 6.7.4.1 for verification of the RCGVS flow capacity. On this basis, DSER Open Item 6.7.1-2 is resolved.

6.7.2 Rapid Depressurization System

The RDS is designed as a manually operated safety-grade system that removes steam or water from the pressurizer through two isolation valves in each of two parallel depressurization lines to the IRWST. The RDS is designed to mitigate the consequences of a beyond-design-basis event such as a total loss of normal and emergency feedwater (TLOFW) event.

The RDS valves are closed during normal operation. These valves are motor-operated and fail in the "as is" position. The valves are designed to be operated during a station blackout (SBO). Thus, the Class 1E dc buses supply electrical power to the motor operators (through dc to ac inverters). The FMEA documented in CESSAR-DC Table 6.7-3 demonstrates that an RDS bleed path can be established in the event of a single failure in the mechanical equipment or battery banks with total SBO.

The design-basis event for determining the size of the RDS bleed valves is a TLOFW event. ABB-CE performed the analysis using a realistic version of the CEFLASH-4AS(REM) code with assumed best estimate decay heat values. The CEFLASH-4AS(REM) is documented in CEN-420P, "SBLOCA Realistic Evaluation Model". The staff determined that use of the realistic version of the CEFLASH-4AS code is acceptable because the RDS is designed to mitigate accidents beyond the design basis. In the analysis of determination of the SDS valve size, ABB-CE did not credit letdown, charging, and pressurizer spray. In the accident scenario, ABB-CE assumed that the initial RCS power and secondary steam are generated at the rated output. The primary and secondary safety valves open at lift pressures of 17.2×10^3 kPa (2500 psia) and 8.27×10^3 kPa (1200 psia), respectively, and the RCPs trip 10 minutes after the event is initiated. In the DSER, the staff stated that ABB-CE should justify that a delay time of 10 minutes to trip RCPs is conservative in calculating the required size for the RDS valves. This was designated as DSER Open Item 6.7.2-1.

In response to DSER Open Item 6.7.2-1, ABB-CE provided in a submittal of November 24, 1992, and CESSAR-DC 6.7.1.2.1.C.2 (Amendment U), a technical basis for operator action time of 10 minutes. ABB-CE followed the guidance in ANSI/ANS-51.1-1983, "American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," and considered a TLOFW event as a plant condition 2 event. According to ANSI/ANS-58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," a plant condition 2 event requires at least 10 minutes from a TLOFW initiation until four RCPs are tripped. The staff has relied on the guidance in ANSI/ANS-58-1984 for approval of TS changes to the Haddam Neck Plant (an October 22, 1990 letter from A. Wang (NRC) to E. Mroczka (CYAPC)). Therefore, the staff concludes that the assumption of operator action time of 10 minutes to trip four RCPs in sizing the RDS valve is acceptable. On this basis, DSER Open Item 6.7.2-1 is resolved.

ABB-CE analyzed two cases (in the response to RAI Q440.22): (1) a TLOFW event with one RDS bleed path available, two SI pumps operable, and immediate operator action to open the RDS bleed path after the primary safety valves (PSVs) open and (2) a TLOFW event with both RDS bleed paths operable, four SI pumps operable, and an operator action delay of 30 minutes to open the RDS paths after the PSVs open. The analysis shows that case 2 is the worst case, which requires larger RDS bleed valves, each of 1.95×10^{-3} m² (0.021 ft²), to comply with the acceptance criterion. This criterion requires the minimum level of mixture water in the reactor coolant to remain two feet above the top of the core through the transient. Since the calculated size of the valve depends heavily on the computer codes used for analysis, ABB-CE thus must confirm the RDS valve capacity in the ITAAC program.

In response to DSER Open Item 6.7.2-2, ABB-CE included an item in the SDS ITAACs to verify the valve flow capacity. Shop tests measuring the steam flow through the RDS valve will be performed. As described in CESSAR-DC Section 6.7.4.1 (Amendment T), the CEFLASH-4AS(REM) will be used to model the performance of the RDS. The computer model of the RDS valves will be initialized to predict the same steam flow rate as the vendor data from shop testing of the RDS valves. The code will then be used in simulating a TLOFW event. To demonstrate the adequacy of the RDS design, the calculated results using the CEFLASH-4AS(REM) will show that the reactor vessel water remains 2 feet above the top of the core during the TLOFW event, which is the event used for determining of the RDS valve sizes. The staff finds that ABB-CE's tests and analysis are adequate for the verification of the RDS valve capacity. On this basis, DSER Open Item 6.7.2-2 is resolved.

Engineered Safety Features

The RDS also performs an important function in mitigating a severe accident. During a core melt, the system would allow the RCS to be depressurized and reduce the possibility of a challenge to the containment, such as from direct containment heating.

In response to the staff's question (RAI Q440.19) on the operator delay time to open RDS valves for severe accidents, ABB-CE proposed allowing the operator delay time of 1.5 hours to open the RDS valves after the PSV lifts for accident mitigation. The severe accident for determining the operator delay time is a total SBO including the failure of the onsite alternate ac power source, the complete loss of secondary-side heat removal capability, and no SI flow. ABB-CE used Revision 16 of the computer code, MAAP3.B, to model the RCS, the steam relief system, and the IRWST. The results demonstrated that a delay of 1.5 hours by the operator will not violate the acceptance criterion that requires the RCS be depressurized from 17.2×10^3 kPa (2,500 psia) to 1,720 kPa (250 psia) before the reactor vessel fails. ABB-CE has confirmed that the 1,720 kPa (250 psia) criterion precludes a direct containment heating challenge. On this basis, DSER Confirmatory Item 6.7.2-1 is resolved (see Section 19.2.3.3.3 on this report for the staff's evaluation for the resolution of this confirmatory item).

In response to the staff's RAI Q440.20, ABB-CE stated that equipment procurement specifications will define the expected environmental parameters to satisfy the requirements for equipment survivability. However, ABB-CE did not submit specific information. The staff asked ABB-CE to submit a discussion of the specific environmental parameters (such as temperature, pressure, moisture and radiation) and to submit the guidance for meeting the equipment survivability requirements during a severe accident. ABB-CE has submitted an evaluation and demonstrated that the RDS would be available in responding to severe accident conditions. On this basis, DSER Open Item 6.7.2-3 is resolved. The staff's evaluation of the RDS valve operability during a severe accident is discussed in Section 19.2.3.3.7 of this report.

In response to the staff's request, ABB-CE proposed a TS for a RDS in CESSAR-DC Section 16.7.18. The TS requires a COL holder to verify every 18 months that all manual isolation valves to pressure instruments in each vent path are in the open position, and to cycle every 18 months the RDS valves in each vent path at least one complete cycle from the control room. The TS also requires a COL holder to verify every 12 hours that the valve position indication in the MCR is correct for all RDS valves, and verify every 7 days that the breaker alignment is correct and the power to the position indication is available. The staff finds that the proposed TS is accept-

able to assure the operability of the RDS valves. On this basis, DSER Confirmatory Item 6.7.1-1 (regarding the TS for RDS) is resolved.

In response to DSER Open Item 6.7.2-4, ABB-CE included the RDS in EOGs for the feed-and-bleed mode of operation. Since the entry conditions for the feed-and-bleed operation are specified and the guidelines specified in PC-6 and HR-4 of the FRG are clearly written, the staff finds that the EOG instructions provide adequate guidelines for the pressure control and heat removal by using the RDS. On this basis, DSER Open Item 6.2.7-4 is resolved. The staff's evaluation of ABB-CE's response to the severe accident mitigation is included in Section 19.2.3.3.7 of this report.

6.7.3 Conclusions

The staff has reviewed the SDS description in CESSAR-DC Section 6.7, the piping and instrumentation drawings, and the response to the staff's review questions. The SDS design is acceptable because it meets GDC 17, 19, 34, and 36 and RG 1.29.

6.8 In-Containment Water Storage System

In CESSAR-DC Section 6.8, ABB-CE presents the design and description of the in-containment water storage system. This system will collect, deliver, and store water and is the heat sink inside the containment during normal operation and accident conditions. It consists of the IRWST, the HVT, the steam relief system (SRS), and the cavity flooding system (CFS).

The IRWST is the safety-grade source of borated water for the SIS, CSS, CFS, and refueling operation. The IRWST is the primary heat sink and collector of any discharge and leakage from the pressurizer safety valve or rapid depressurization valve. SRP Section 5.4.11 includes the guidance for the staff to review the pressurizer relief tank (PRT) system in current plants. However, the PRT in current PWR plants is a non-safety-related system. Thus, the staff reviews the PRT primarily to ensure that it will operate consistently with transient analyses of related systems and that failure or malfunction of the system could not damage affect essential systems or components. The IRWST in the System 80+ design is the safety-grade source of water after an accident. Therefore, the staff reviewed the IRWST using the applicable sections of SRP Section 5.4.11, SRP Section 6.2.2, and RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems." These additional requirements are derived from the functional requirements of the IRWST design.

The HVT collects and stores water during normal and accident conditions. The SRS will transport steam or water relieved through the PSVs and rapid depressurization valves (RDVs) to the IRWST, and dissipates its energy in the IRWST water. The CFS floods the reactor cavity during a severe accident, and has been reviewed as part of the severe accident. This evaluation is given in Section 19.2 of this report. During normal operation, the SCS heat exchangers can cool the IRWST water. During a LOCA or secondary side pipe break, the containment spray system heat exchangers cools the IRWST. During the "bleed" phase of RCS rapid depressurization, the IRWST may be cooled by the SCS or the containment spray system heat exchangers.

The IRWST is located inside the spherical steel containment vessel. Its boundaries are defined by the available space in the lower regions of the reinforced-concrete internal structure. The upper concrete dish of the internal structure is the bottom of the IRWST. The top is the concrete slab. The outer radial wall is the concrete crane wall and the inner radial wall is the semicircular wall of the HVT and the primary shield wall. The IRWST has total volume of 2.066×10^6 L (545,800 gal) and $1,676 \text{ m}^3$ (59,200 ft^3) feet of free space.

In Amendment U of CESSAR-DC Section 6.8.2.2, ABB-CE states that overpressure protection for the IRWST is provided by four pressure relief dampers in the IRWST cover. These dampers, which are self-actuated by IRWST pressure, are provided for hydrogen venting during a severe accident and will accommodate the consequences of a DBA steam release and rapid depressurization valve (RDV) actuation, assuming no cooling of the IRWST is available. The dampers are also actuated by a vacuum in the IRWST. These dampers are bi-directional and spring-loaded. The low-volume containment purge subsystem vents the IRWST during normal fill of the IRWST.

If one or more PSVs or RDVs actuate, steam or water will flow through the main discharge lines to distribution headers in the IRWST and into the sparger heads. High-pressure jets of steam will be injected into the IRWST, where it will be condensed and mixed with the IRWST water. The IRWST, in this case, will function similarly to the suppression pool in the boiling-water-reactor plants. However, in earlier versions of the CESSAR-DC, ABB-CE did not address the hydrodynamic loads to the IRWST and SRS. In the DSER, the staff stated that ABB-CE must submit additional information to address this concern. This was designated as DSER Open Item 6.8-1. In Amendment N of CESSAR-DC, ABB-CE provided the requested information in Section 6.7.6, "Hydrodynamic Loads on the Safety Depressurization System (SDS)," and Section 6.8.4, "Hydrodynamic Loads on the Incontainment

Refueling Water Storage Tank." On the basis of its analyses, ABB-CE concluded that the loads on the SDS piping and IRWST are within the design capability of piping and supports for safety relief valve piping and the design capability of the IRWST structural elements. The staff concurs with this analysis and, on this basis, DSER Open Item 6.8-1 is resolved.

The in-containment water storage system is Safety Class 2. Therefore, it will be manufactured, and tested in accordance with the rules of ASME Boiler and Pressure Vessel Code, Section III.

The piping and instrumentation diagrams for the IRWST include temperature and level indication, but no pressure indication for the tank. Responding to the staff's RAI Q440.69, ABB-CE stated that the level instruments that indicate the IRWST level provide high and low level alarms in the control room. The diagrams reference instrumentation requirements in CESSAR-DC Section 7.4.1.3, which ABB-CE had not submitted.

Responding to Q410.102a, ABB-CE submitted a description of the level, pressure, and temperature indication, a revised drawing without the reference to Section 7.4.1.3, and a new Section 6.8.3 to define the instrumentation. This completes the information required and, therefore, is acceptable.

Responding to Q440.69, ABB-CE stated that the IRWST is designed to comply with SRP Section 6.2.2 and RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems." The staff concludes that the IRWST design and analysis comply with the requirements of RG 1.82 and SRP Section 6.2.2 as discussed in Section 6.2.2.

On the basis of this evaluation, the staff finds the in-containment water storage system acceptable in meeting the applicable guidance of SRP Section 5.4.11, SRP Section 6.2.2, and RG 1.82.

7 INSTRUMENTATION AND CONTROLS

7.1 Identification of Safety Systems, Review Criteria, and Design Processes

Section 7.1 of the ABB-CE Standard Safety Analysis Report for Design Certification (CESSAR-DC) describes the instrumentation and control (I&C) systems in the ABB-CE System 80+ advanced light water reactor (ALWR) design that are important to safety, and the acceptance criteria and guidelines applicable to each of these systems. CESSAR-DC also discusses those I&C systems important to safety that are identical to the CE System 80 (Palo Verde) design and have already been reviewed and approved by the Commission.

7.1.1 Identification of Safety-Related Systems

The staff reviewed the ABB-CE choices of I&C systems important to safety to confirm consistency with the design bases for safety systems as discussed in other sections of CESSAR-DC, particularly Chapters 5, 6, 8, 9, 10, and 15. These I&C systems are summarized in this section. The following sections describe the results of the NRC's review and evaluation.

The ABB-CE System 80+ plant protection system (PPS) consists of the reactor protective system (RPS) and the engineered safety features actuation system (ESFAS). The RPS initiates a reactor trip if the reactor conditions approach prescribed safety limits. The ESFAS actuates the engineered safety features (ESF) systems. The RPS is discussed in Section 7.2, and the ESFAS is discussed in Section 7.3.

The reactor trip system (RTS) includes the RPS portion of the PPS, the reactor trip switchgear system (RTSS), and the components that perform the protective action either automatically or manually. The RTS initiates a reactor trip based on various nuclear steam supply system (NSSS) parameters and the containment pressure.

The ESF systems include the ESFAS and components that perform protective and mitigative actions after receiving a signal from the ESFAS or the operator.

The ESF systems follow:

- containment isolation
- main steam isolation
- safety injection (SI)
- emergency feedwater (EFW)
- containment spray
- safety depressurization
- support

The actuation logic of each ESF system is similar to the logic for the RPS and is contained in the engineered safety features component control system (ESF-CCS). The design of this system is described in Section 7.3. The ESFAS in the ABB-CE System 80+ differs from the CE System 80 (Palo Verde) design in that it:

- Does not include the recirculation actuation signal because the in-containment refueling water storage tank (IRWST) was added.
- Includes an emergency feedwater actuation signal (EFAS) initiation logic that is not required to automatically detect and isolate a ruptured steam generator. A cavitating venturi restricts the maximum flow to each steam generator to protect the EFW pumps from damage caused by excessive runout, and to allow the operator 30 minutes to regulate or terminate EFW flow to prevent reactor coolant system (RCS) overcooling, steam generator overfill, or containment overpressurization.

The systems required for safe shutdown are those systems essential for pressure and reactivity control, coolant inventory makeup, and residual heat removal once the reactor has been brought to a subcritical condition. These systems are organized according to the following shutdown modes:

- (1) Hot shutdown systems are those required for maintenance of the primary system at or near operating temperature and pressure. They comprise the following:

- emergency diesel generator (EDG)
- emergency diesel generator fuel storage and transfer
- emergency power storage
- emergency onsite power distribution
- safety injection
- emergency feedwater
- atmospheric steam dump
- safety depressurization
- station service water
- component cooling water
- heating, ventilation, and air conditioning (HVAC)

The design includes remote shutdown panel (RSP) equipment and systems to allow emergency shutdown from outside the control room.

- (2) Cold shutdown systems are those required to cool down and maintain the primary system at, or near, ambient conditions. These systems consist of all

Instrumentation and Controls

the hot-shutdown systems listed above, and the shutdown cooling system (SCS).

These systems are discussed in Section 7.4.

The safety-related display instruments give the operator information concerning plant operating conditions and information necessary to perform manual safety functions. Safety-related displays indicate the status of

- plant processes
- reactor trip system
- engineered safety features actuation system
- control element assembly (CEA) position
- postaccident conditions
- ESF systems performance and availability
- critical functions

The safety-related display information systems are discussed in Section 7.5.

Other systems required for safety include the interlocks to prevent overpressurization of the SCS and to ensure SI availability. These are listed below and described in Section 7.6.

- shutdown cooling system suction line isolation valve interlocks
- safety injection tank (SIT) isolation valve interlocks

Control systems whose functions are not essential for the safety of the plant are discussed in Section 7.7 and include the following:

- reactivity control
- pressurizer pressure and level control
- megawatt demand setter (MDS)
- feedwater control
- steam bypass control
- reactor power cutback
- boron control
- in-core instrumentation
- ex-core neutron flux monitoring
- boron dilution alarm
- alternate protection
- process component control
- control and monitoring systems sensed parameters
- severe-accident

7.1.2 Review Criteria

The staff performed this review using acceptance criteria in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," standard review plan (SRP) and 10 CFR

Part 52. The Commission also gave review guidance in staff requirements memoranda (SRM) dated June 26, 1990, and February 15, 1991, pertaining to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," and SECY-90-377, "Requirements for Design Certification under 10 CFR Part 52," respectively.

The SRP does not address design certification, or the newer digital technology implemented in the ABB-CE System 80+ I&C systems. Therefore, as stated in SECY-91-292, "Digital Computer Systems for Advanced Light Water Reactors," the staff developed the necessary acceptance criteria using applicable national and international standards. The additional acceptance criteria and conformance to the SRP are discussed in the applicable sections of this report.

Many of the standards referenced by ABB-CE are revisions of the standards referenced by the NRC staff in the SRP. The staff reviewed these standards to determine their applicability to the NRC design review criteria. The standards referenced by ABB-CE envelop the scope of the standards and guidelines used by the staff. Consequently, the staff finds acceptable for this design certification the standards referenced by ABB-CE. The staff applied the referenced standards as acceptance criteria for the System 80+ design.

ABB-CE did not complete the hardware and software design for the System 80+ digital I&C systems. Therefore, the staff used the two-part approach given in SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," to reach its safety finding for design certification. In reviewing the I&C systems, the staff began by performing a detailed functional review of block diagrams of the I&C system architecture to ensure ABB-CE appropriately implemented the Commission's requirements for postulated single failures, common-mode failures (CMFs), signal isolation, and other aspects of the staff's review that are typical for any safety-related I&C system including analog control systems such as those in current operating nuclear plants. This review confirmed that the detailed functional requirements for the instrumentation and control systems are met.

ABB-CE must obtain NRC approval before changing safety analysis report (SAR) commitments that affect the technical design of the I&C systems, including the design process, design implementation, and the NRC staff review as described in this report. The majority of the SAR material may be revised through a "50.59-like" process as described in the rule certifying the design. This report also specifies those areas that require staff review before being changed. The areas of the System 80+ I&C design that

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the staff has determined must be submitted for review of proposed changes include computer design (hardware and software), multiplexor design, setpoint methodology, electromagnetic compatibility, and commercial grade item dedication. These areas are designated in the final safety evaluation report with the following statement:

Any changes to this commitment would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

In the second part of the staff's approach to reach a certification safety finding, the staff verifies that the implementation of the digital control systems meets the system functional requirements. This relies upon a formal design implementation process with a phased inspection, test, analysis, and acceptance criteria (ITAAC) program for design development within certain predefined constraints and limits.

The System 80+ includes a digital I&C system to perform I&C functions. ABB-CE used a structured design process to assign functional system (RPS, ESF-CCS, etc.) requirements to specific computer hardware and software components to accomplish the various tasks. ABB-CE submitted the implementation plans for this process, as described in Section 7.1.4.

The design certification material described the design process to be followed for hardware and software development, design commitments, and ITAAC. The ITAAC for systems that use computer hardware and software differ from other system ITAAC in that they include design acceptance criteria (DAC) for the certified design process as discussed in SECY-92-053. These ITAAC describe attributes of the process for developing software and attributes of the final software product. The staff will audit the compliance with ITAACs at various points during the digital system design process.

7.1.3 Diversity and Defense-in-Depth

The System 80+ I&C systems are microprocessor-based systems that share processing functions (software) and process equipment (hardware). Therefore, a hardware design error, a software design error, or a software programming error may cause redundant equipment to fail. The staff is concerned that the use of digital computer technology could result in safety-significant CMFs. CMFs could both defeat the redundancy achieved by the hardware architectural structure and result in the loss of more than

one echelon of defense-in-depth provided by the I&C system. The two principal factors for defense against CMFs are quality and diversity. Quality in the design process is addressed in Section 7.1.4. For the software portion of safety-related I&C systems, the staff position on defense-in-depth and diversity for ALWRs, as approved by the Commission in an SRM dated July 21, 1993, in response to SECY-93-087, is as follows:

- (1) ABB-CE shall assess the defense-in-depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to CMFs have been adequately addressed.
- (2) In performing the assessment, the vendor or applicant shall analyze each postulated event that is in the accident analysis section of the SAR using best-estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events.
- (3) If a postulated CMF could disable a safety function, then a diverse means, with a documented bases that the diverse means is unlikely to be subject to the same CMF, shall be required to perform either the same function or a different function. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.
- (4) A set of displays and controls located in the main control room (MCR) shall be provided for system-level actuation and control of critical safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in items 1 and 3.

The staff's proposed applicable regulation for digital I&C diversity is as follows:

Digital instrumentation and control systems provided for the standard design must include:

- (1) an assessment of the defense-in-depth and diversity of the instrumentation and control systems;
- (2) a demonstration of adequate defense against common-mode failures; and
- (3) provisions for independent backup manual controls and displays for critical safety functions in the control room.

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ABB-CE revised the original I&C system design to incorporate features that are consistent with the Commission approved position. The revised system design is discussed in Section 7.2.

7.1.4 Software and Hardware Development

The System 80+ uses microprocessor-based digital equipment to perform the safety-related and non-safety-related I&C system functions. The safety-grade classification of the software used in these systems must be consistent with that of the system hardware. Consequently, the safety-related requirements of the System 80+ software-based systems must be implemented using high-integrity software.

To satisfy the requirements for high-integrity software, the software functions of the System 80+ safety system equipment and all other safety-related software must conform to regulatory requirements and industry standards governing these activities. This conformance requirement applies to software designed specifically for the System 80+ and software originally designed for other applications and procured for use in the System 80+ design. The non-safety-related software should also be developed using a structured approach similar to that of the safety-related software, but not necessarily with strict conformance to the criteria required of the safety-related software.

Many of the requirements for safety-related software include aspects that involve the complete hardware and software design because software is integrated into hardware components. ABB-CE need not complete the final design details of the System 80+ I&C system before the COL is issued. The rapid changes in this technology dictate that the certified design description and ITAAC do not prescribe a design and design methodology that would be obsolete at the time of construction.

The design certification includes a description of the design process. The ITAAC that address software differ from other system ITAAC in that the acceptance criteria describe attributes of the process to be used to develop the software and specific attributes of the final software product.

ABB-CE submitted a software development plan (SDP), NPX80-SQP-0101.0, "Nuplex 80+ Software Program Manual," which describes the complete software development process including hardware integration. This plan has been referenced in the SAR.

The staff reviewed ABB-CE's software development process based on guidance and criteria in the following standards:

1. American Nuclear Society (ANS) American National Standards Institute (ANSI)/Institute of Electrical and Electronics Engineers (IEEE) Std 7-4.3.2-1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations"
2. ANSI/IEEE Std 730-1989, "IEEE Standard for Software Quality Assurance Plans"
3. ANSI/IEEE Std 828-1983, "IEEE Standard for Software Configuration Management Plans"
4. ANSI/IEEE Std 1012-1986, "IEEE Standard for Software Verification and Validation"
5. ANSI/IEEE Std 1042-1987, "IEEE Guide to Software Configuration Management"
6. ANSI/IEEE Std 1058.1-1987, "IEEE Standard for Software Project Management Plans"
7. American Society of Mechanical Engineers (ASME) NQA-2a-1990 Addenda, Part 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications"

ABB-CE's SDP consists of a software quality assurance (SQA) plan, a software configuration management (SCM) plan, a software verification and validation (V&V) plan, and an operations and maintenance (O&M) plan. The elements of a software management plan are incorporated into the SQA plan, the V&V plan, and the SCM plan. ABB-CE will prepare the integration plan, installation plan, and training plan during the remainder of the planning stage. Although the software safety plan (SSP) was submitted independently of the SDP, the SDP includes the elements of the SSP.

7.1.4.1 Software Quality Assurance Plan

The SQA plan describes the software-specific activities that are to be performed and controlled in addition to the approved quality assurance plan (in accordance with 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants") for the total System 80+ design. The SQA plan establishes the criteria under which the other SDPs will be generated. The staff reviewed the SQA plan using ANSI/IEEE Std 730-1989, which states that a SQA plan shall include the following sections:

1. purpose
2. reference document
3. management
4. documentation

5. standards, practices, conventions, and metrics
6. reviews and audits
7. test
8. problem reporting and corrective action
9. tools, techniques, and methodologies
10. code control
11. media control
12. supplier control
13. records collection, maintenance, and retention
14. training
15. risk management

The staff reviewed the SQA plan and forwarded comments to ABB-CE for resolution. ABB-CE resolved the staff's concerns regarding format and scope and reissued the SQA plan as part of the SDP. The staff concludes that the SQA plan, in conjunction with the SSP, generally conforms to the guidelines of ANSI/IEEE Std 730-1989 and ASME NQA-2a-1990 addenda Part 2.7 and adequately addresses the above topics, and is therefore, acceptable.

7.1.4.2 Configuration Management Plan

The configuration management plan (CMP) will enable the COL applicant to identify software products, control and implement changes, and record and report change implementation status. The staff reviewed ABB-CE's CMP using the criteria of IEEE 828-1983, which states that software CMPs shall address the following topics:

1. management organization, responsibilities, interfaces, implementation, policies, directives, and procedures
2. SCM activities concerning configuration identification, control, status accounting, audits, and reviews
3. tools, techniques, and methodologies
4. supplier control
5. records collection and retention

ABB-CE's CMP describes the general organizational structure that influences the configuration of the software during the development and the operation and maintenance phases and the independence of the elements responsible for system software configuration management functions from those responsible for verification and validation activities related to configuration management. The CMP has requirements for configuration management responsibilities including configuration identification, software change request, software change authorization, module and unit release history, baselines, and backups. The CMP describes the software configuration management activities

related to the software project baselines, the configuration change control authority and management, methods of access control, and the configuration status control log maintenance. Based on the above, the staff concludes that the CMP is acceptable. System specific configuration management plans that reflect the specific methods of managing the software configurations will be developed as part of the SQA plan discussed above.

7.1.4.3 Software Verification and Validation Plan

The software V&V plan will enable the COL applicant to evaluate whether software through its life cycle to ensure that the software system requirements are accurately translated into the final software product. Each software module will be verified by an organization that is independent of the organization that developed the software module. The staff reviewed ABB-CE's V&V plan using the criteria of IEEE 1012-1986, which states that software V&V plans shall define the following activities for each phase of the software life cycle:

1. V&V tasks
2. methods and criteria for V&V
3. V&V process inputs and outputs
4. schedules for performing V&V tasks
5. resources
6. risks and assumptions
7. roles and responsibilities

ABB-CE referenced ANSI/IEEE-ANS-7-4.3.2-1982 as the standard for designing, verifying, implementing, and validating the software in the safety-related systems. This standard is endorsed by Regulatory Guide (RG) 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants." The staff reviewed ABB-CE's computer system development process using industry and international standards as review criteria.

ANSI/IEEE Std 1012-1986 addresses detailed software V&V activities performed during each phase of the software life cycle. The software lifecycle may be described as follows:

1. concepts [planning]
2. requirements
3. design
4. implementation
5. test
6. installation and checkout
7. operation and maintenance

Figure 7.1. Digital system development activities

These stages are based on generic activities that represent software and system life cycle processes, and encompass expert opinions, national standards, and information from ongoing efforts to revise national and international standards for computer systems in nuclear power plants. Figure 7.1 depicts documents and activities from this software development process.

ABB-CE defines the System 80+ software lifecycle as consisting of the following stages:

1. requirements analysis
2. design
3. implementation or coding
4. test
5. installation and checkout
6. operation and maintenance
7. retirement

ABB-CE stated that these computer development stages or phases encompass those described in industry standards. ABB-CE incorporated the planning stage (shown in Figure 7.1) into the requirements development stage. The staff concludes that these stages conform to the encompass the software lifecycle stages described in ANSI/IEEE Std 1012-1986 and are acceptable for the development of the safety-related and non-safety-related computer systems.

ABB-CE's V&V plan acceptably addresses each of the above software life cycle phases. The staff concludes that the V&V plan conforms to those in RG 1.152 (ANSI/IEEE-ANS-7-4.3.2-1982) and ANSI/IEEE Std 1012-1986 and is acceptable.

7.1.4.4 Software Operation and Maintenance Plan

The software operation and maintenance (O&M) plan includes the procedures required to ensure that the digital system will be operated correctly, and the quality of the software will be maintained. ABB-CE's O&M plan specifies the organizational requirements and error-reporting requirements that the COL holder shall follow during the operation phase and the retirement phase of the software life cycle. The staff reviewed the O&M plan to evaluate the processes for reporting errors and maintaining the software after ABB-CE releases the product to the COL holder. ABB-CE committed to maintain the safety-related systems under the requirements of 10 CFR Part 50 and 10 CFR Part 21. The staff finds this commitment to be acceptable. The O&M plan includes error-reporting requirements and forms to ensure that digital system errors are reported in a manner that is consistent with the configuration management guidelines in the CM plan section of the SDP. The staff concludes that the O&M plan includes the activities necessary to maintain the Nuplex 80+

software, the remove latent errors, to respond to new or revised requirements, and to adapt the software to changes in operating environment and is acceptable.

7.1.4.5 Software Management Plan

The software management plan establishes the organization and authority structure for the design, the procedures to be used, and the relationships between major activities. The staff reviewed the SDP to evaluate the adequacy of the software management requirements. The staff concludes that the management structure in the SDP provides adequate project oversight, control, reporting, review, and assessment and is acceptable.

7.1.4.6 Software Program Manual

The staff determined that the following topics of the software program manual (ABB-CE's SDP) cannot be changed without prior NRC approval because these sections contain basic information relied on by the staff for its safety determination:

1. Introduction
 - (a) Purpose
 - (b) Scope
 - (c) Overview
2. Software Quality Assurance Plan
 - (a) Purpose
 - (b) Management
 - (c) Documentation
 - (d) Review Requirements
 - (e) Problem Reporting and Corrective Action
 - (f) Code Control
3. Software Verification and Validation Plan
 - (a) Purpose
 - (b) Overview of V&V
 - (c) System Validation Testing
4. Software Configuration Management Plan
 - (a) Introduction
 - (b) Management
5. Software Operation and Maintenance Plan
 - (a) Purpose
6. Documentation
 - (a) General Requirements
 - (b) Functional Requirements Documents
 - (c) Software Design Requirements
 - (d) Software Design Description
 - (e) Software Verification and Validation Documentation

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(f) Computer Code Certificate

7. Problem Reporting and Corrective Action

- (a) Classification System
- (b) Problem Reporting
- (c) Corrective Action

The COL applicant must first obtain NRC approval before making any changes to these topics that would involve an unreviewed safety question, except for editorial changes and layout, and therefore, will require NRC review and acceptance prior to implementation. Any requested changes to these commitments shall be specifically described in the COL application or submitted for license amendment after the COL is issued.

7.1.4.7 Software Safety Plan

The staff reviewed ABB-CE's SSP, NPX80-IC-QP790-02, "Nuplex 80+ Software Safety Plan Description," using the guidance in IEEE Draft Standard P1228, "Software Safety Plans." ABB-CE's SSP addresses the topics described in IEEE P1228. ABB-CE's SSP addresses the topics presented in IEEE P1228 through references to the SDP, and references organizational structure and responsibilities, resources, methods of accomplishment, depth of effort, and integration of system safety with other program engineering and management activities. The hazards evaluations required by the SSP will be documented in the V&V documentation. The staff concludes that the SSP generally follows the guidance in IEEE P1228 and is acceptable.

7.1.4.8 Software Development Process

The process for developing and implementing software complies with the regulatory requirements and industry standards governing those activities. The process meets both the requirements of 10 CFR Part 50 for I&C systems (50.55a(h) and General Design Criteria (GDC) 1, 2, 4, 13, 19, 20, 21, 22, 23, 24, 25, and 29 in Appendix A to Part 50) and the requirements in the design certification material. The safety-related SQA program will be implemented by the vendor and evaluated by the NRC in accord with 10 CFR Part 50 (Appendix B).

ABB-CE's SDP referenced in the SAR describes software development process and audit activities, which include the plans described in the previous sections. The staff reviewed ABB-CE's SDP using the standards listed above in Section 7.1.4 and confirmed that the SDP acceptably meets the requirements and guidelines in the standards that address the planning stage processes shown in Figure 7.1.

The COL applicant will document its actions in the remaining phases of the software development process. The staff will review these documents during the ITAAC process audits as the software systems are developed and implemented.

The computer system development process shown in Figure 7.1 illustrates the relationship between the implementation of the software-based systems and the NRC staff's ITAAC audits.

The COL applicant will submit the documents listed in Figure 7.1 for staff audit through each life cycle stage of the development process. Following each audit, the NRC staff will issue a report documenting the completion of that ITAAC stage (inspection, test, and analyses) and list open items that will require resolution. Significant unresolved items could prevent the NRC staff from confirming completion of the ITAAC. Resolution of these items may result in changes to information in software design documents and materials that have been the basis for certain ITAAC confirmations made during previous NRC audits or conformance reviews. These ITAAC may require reconfirmation after the documents and materials are changed.

At each stage of the life cycle, the NRC staff will conduct inspections and audits to confirm that the design is in accordance with the certified design description and with the more detailed design commitments made in the previous stages. At each stage, the COL applicant will submit a description of the next stage of design in enough detail that the NRC staff can objectively determine conformance with the certified design requirements when that stage is completed and submitted for staff audit. The detailed description will include the accepted industry standards and regulatory requirements that apply when the software design is started. This phased process will continue until the COL applicant completes the ITAAC steps for all the safety-related software.

The certified design description will apply to the System 80+ after the design is complete, the ITAACs have been completed, and plant operation has started. Any software changed or added after plant startup must be of the same or higher quality. It is not possible to predict all software design developments during the combined lifetime of the design certification and plant operation. Consequently, the COL holder will use the certified design process to develop software changes and submit the software to the NRC for review, or use specific accepted methods not described in CESSAR-DC that will produce software of the same or higher quality than current design practices. The COL holder will be required to use an approved software change procedure for the operation stage of the life cycle.

Any changes to these requirements would involve an unreviewed safety question, and therefore, will require NRC review and acceptance before implementation. Any requested changes to these requirements shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

7.1.4.9 Commercial Dedication

ABB-CE committed to procure commercial, off-the-shelf (COTS) software through its commercial dedication program for implementation in the safety-related I&C systems. ABB-CE submitted a document describing the commercial dedication program document to the staff for review (NPX80-QPS-0401.1, "Requirements for the Supply of Commercial Digital Computer Hardware and Software To Be Used in Nuplex 80+ Safety Systems"). This document is referenced in ABB-CE's SDP. ABB-CE will employ commercial equipment used in safety-related applications through a commercial dedication process that the staff reviewed and found acceptable in accord with the guidance in Electric Power Research Institute (EPRI) NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Application (NCIG-07)."

The commercial-grade computer hardware and software items will be incorporated in safety-related systems through a process that requires:

1. requirements for supplier design control, configuration management, problem reporting, and change controls
2. reviews of product performance
3. receipt acceptance of the commercial-grade item from the supplier
4. final acceptance, based on equipment qualification and software validation in the integrated system

These requirements have been placed into the system ITAACs. Any changes to these commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance before implementation. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

The first aspect of commercial dedication is the use of well-developed operating systems in plant-specific digital systems. ABB-CE states that proven technology will be employed in the design and development of the System 80+ I&C systems. ABB-CE's criteria for component service is that the component has been in service for at least 3,000 operating years and in the field at least one

calendar year. The staff finds this requirement acceptable based on similar requirements in the EPRI Utility Requirements Document (URD) for ALWRs. The stated goal, to which both the staff and ABB-CE agrees, is to use the best available technology without using unproven designs.

The COL applicant need not perform V&V of a COTS operating system as part of the software design implementation process but must ensure that the operating system has been developed under strict guidelines and has the quality necessary for a safety system.

The second aspect of commercial dedication is the use of a complete component, such as a programmable logic controller, for which most of the software was developed before the decision to use it in a nuclear application. The COL applicant must verify that the equipment selected is of sufficiently high quality for use in a safety system. The COL applicant need not repeat the V&V activities, but must verify that the original equipment designer followed equivalent criteria.

Commercial dedication includes the qualification of the automated tools and design support software. The I&C system developer must verify that the tools are accurate. The staff expects the COL applicant to verify the quality of the tools used in the design.

The staff expressed concern that COTS suppliers inform the COL applicant of errors discovered in the suppliers' tools or software to the end user in a manner similar to the 10 CFR Part 21 defect reporting required for safety-related (Class 1E) equipment vendors.

Any changes to the software development commitments for commercial dedication would involve an unreviewed safety question and, therefore, require NRC review and acceptance before implementation. Any requested changes to this commitment shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

7.2 Reactor Protective System

7.2.1 Discussion

The PPS consists of the equipment necessary to monitor selected plant conditions and to actuate a reactor trip and ESF components. The two major subsystems of the PPS are the RPS and the ESFAS. The RPS is described in this section. The ESFAS is described in Section 7.3.

The RPS will rapidly shut down the reactor when certain plant conditions approach safety system setpoints. The

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RPS is segregated into four completely independent channels consisting of sensors and transmitters, signal conditioning, bistable logic, core protection calculators (CPCs), local coincidence logic (LCL), and initiating relays. Safety-related, independent sensors in each channel will be continuously monitored to provide signals that are processed by the PPS in the following sequence of functions:

1. An auxiliary process cabinet (APC) in each channel receives that channel's sensor signals and converts the sensor signals to 0-10 V (analog).
2. The APC transmits the converted analog signals by discrete conductors to one or both bistable processors and/or the CPC within the same channel. The bistable processors utilize Programmable Logic Controller (PLC) technology. The CPC uses 32 bit minicomputer technology.
3. The bistable PLCs first convert the analog input signals to discrete digital signals and then compare the signals to the appropriate setpoints. The resulting signal is a binary value: trip or no trip. The bistable PLCs also transmit the digitized signals to the discrete indication and alarm system (DIAS) and the data processing system (DPS) by fiber-optic data link for processing and display. In the initial CESSAR-DC, ABB-CE did not describe the hardware and software design of the bistable PLCs in sufficient detail (DSER Open Item 7.2.1-1). ABB-CE later submitted the following discussion.

Two bistable processors in each channel add reliability to each channel bistable function. Each bistable processor is assigned process measurements for comparison based on an analysis of transients in relation to the mitigating process. When multiple process measurements are available for monitoring a transient, they are assigned to different bistable processors. The design integrates various system components, features, and functions into a microprocessor-based unit that performs the entire trip function.

As stated above, analog input signals are directed to analog input modules (AIMs) within the bistable processors, where the signals are converted from analog form to digital form. The AIMs include self-test and automatic calibration features to eliminate the need for periodic calibration of inputs. Calibration of each bistable input is performed against a precision reference voltage source contained in each module. This reference voltage source requires an annual calibration or as required by the manufacturer. Drift

between calibrations is detected by comparing across channels. Common-mode drift is detected within the DPS by comparing validated values from the DIAS for postaccident monitoring (DIAS-P) and DIAS for normal monitoring (DIAS-N) systems.

Digitized analog values are automatically reported to the bistable central processing unit (CPU) during each PLC scan cycle. Within the bistable CPU, a comparator algorithm determines the pretrip and trip output states. Each output state is determined by comparing the digitized process (from the analog-digital (A/D) converter) to the setpoint (pretrip and trip) from the setpoint algorithm.

The PPS software is deterministic (i.e., repetitive and non-interrupt driven) to ensure predictable system performance and response under all conditions.

Software is divided into two major categories: operating system software and application software. Operating system software consists of the PLC processor operating system, input/output (I/O) handling, communications, and equipment self-test software. Application software is the implementation-specific code that will be developed during the PPS design process.

The PLC manufacturer will write the operating system software code, which will reside in the permanent memory of the PLC processors. ABB-CE qualifies the operating system software by auditing the vendor in accordance with ABB-CE's commercial-grade item dedication program. ABB-CE will validate the software through extensive testing for the intended application in accordance with ABB-CE's SDP as discussed in Section 7.1. After validation, configuration controls will be maintained by ABB-CE for the operating system software in accordance with ABB-CE's CMP.

Application software code resides in permanent memory in the PLC processors, and is separate from the PLC operating system software. This code is written using PLC relay ladder logic language. The individual application software functions are developed as separate software modules. The software modules are divided into trip initiation functions, automatic application software test functions, and status reporting functions.

Each application software module will be verified by an organization independent of the organization that develops the software modules, in accordance with the verification requirements described in ABB-CE's SDP. The purpose of the verification is to ensure that the functional requirements have been correctly implement-

ed in the software modules. After testing, the software modules will be integrated and the independent review process and validation process will be repeated for integrated system operations. This process is acceptably described in ABB-CE's SDP, as discussed in Section 7.1.

The staff finds ABB-CE's description of the bistable PLCs acceptable since the hardware design meets the requirements of IEEE Std 279-1971 (10 CFR 50.55a(h)) regarding channel integrity, channel independence, and redundancy, and the software design process conforms to the requirements described in Section 7.1. On this basis, DSER Open Item 7.2.1-1 is resolved.

4. Two control element assembly calculators (CEACs) in the CPCs prevent CEA misalignments; each monitors one of two redundant, safety-grade CEA position reed switches through fiber-optic data links. The CEAs are arranged into control groups that are controlled as subgroups of CEAs. The subgroups will move together as a control group and will indicate the same CEA group position. If a CEA position deviates from its subgroup position, the CEACs will activate alarms in the DIAS and the DPS, and transmit penalty factors through fiber-optic data links to the trip logic calculators (TLCs) in all four CPC channels.
5. The TLC in each CPC channel monitors the position of one CEA in each CEA subgroup assigned to that channel as a measure of subgroup CEA position. Deviations between the CEAC's position and the TLC position produces control actions to mitigate the event, and, if necessary, reduce the margins-to-trip for low departure from nucleate boiling ratio (DNBR) and high local power density (LPD).

The TLCs use the most conservative insertion deviation penalty factors from the two CEACs to generate a CEA motion inhibit (CMI) signal with one-out-of-two logic for each channel. Each of the four TLC channels sends a CMI signal to the control element drive mechanism control system (CEDMCS) to block rod motion and thereby prevent further CEA deviations. The CEDMCS executes the rod block on coincidence of two-out-of-four CMI signals from the TLCs.

The integrated actions of the CEACs, TLCs, and CEDMCS generate a CMI for withdrawal deviations in the same manner as for insertion deviations. The CEDMCS also prevents withdrawal deviations through its CEA position monitoring, group interlocks, and a self-generated CMI.

6. While the CMI logic is being executed, the TLCs apply the most conservative insertion penalty factor to the DNBR and the LPD calculations. If the results of the calculations indicate a pretrip condition, each TLC will send a reactor power cutback (RPC) signal to the CEDMCS, which then actuates gravity insertion of selected CEAs using two-out-of-four actuation logic.

The RPC prevents a DNBR or LPD trip. The TLCs use the most conservative insertion penalty factor in the DNBR and LPD trip algorithms. If the RPC does not result in sufficient thermal margin, a DNBR trip or a LPD trip, or both will be generated.

If a CEAC is out of service, the TLCs will use the available CEAC penalty factors to generate the CMI, RPC, and reactor trip signals. The CEDMCS also initiates CMI and RPC signals when a CEACs is out of service.

7. The results of the CPC calculations are compared to the corresponding setpoints for low DNBR and high LPD. The results of the comparisons are binary values, trip or no trip, which are transmitted to the channel's LCL PLCs (two per channel).
8. The signals sent to the other channel redundant LCLs are transmitted by fiber-optic data link to ensure signal isolation.
9. A channel parameter may be bypassed for testing or maintenance from the local operator's module in the channel's PPS cabinet or from the PPS channel operator's module in the MCR. The bypass information is sent by fiber-optic data link to the interface and test processors (ITPs) in the four PPS channels. The ITP receives the bypass signal from the MCR or the channel PPS cabinet and receives the bypass status from the other three channels through fiber-optic data link.
10. The LCL PLCs in the four channels use the bypassed parameter status information to process the signals from the remaining channels in a two-out-of-four logic for no channels bypassed or a two-out-of-three logic for one channel bypassed. Once a parameter is bypassed, the interlock logic in the LCLs prevents another channel from also bypassing that signal. If the same parameter in another channel must be taken out of service for maintenance, that channel parameter is placed in trip status, which places the remaining parameter in a one-out-of-two trip state.

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11. Upon detecting a two-out-of-four trip state (or two-out-of-three trip state when a channel is bypassed), the channel's LCL PLCs send the resulting trip signals through the initiation relays by discrete conductors to the channel's dedicated undervoltage trip circuit breaker and shunt trip circuit breaker.
12. The appropriate trip signals also pass through hardwired data link to the ESF initiation relays, which transmit signals through fiber-optic data link to each of the ESF LCL PLCs that process the corresponding trip (e.g., CSAS, SIAS, MSIS).
13. The coincidence signals are used in actuating the RTSS or ESF-CCS.
14. An ITP in each channel of the RPS monitors, tests, and controls the operational state of the bypasses. The ITP also provides fiber-optically isolated RPS channel status and test results information to the DPS and the DIAS.
15. The RPS trips the reactor on the following conditions (the trip processor is in parentheses):
 - variable overpower (PLC)
 - high logarithmic power level (PLC)
 - high local power density (CPC)
 - low departure from nucleate boiling ratio (CPC)
 - high pressurizer pressure (PLC)
 - low pressurizer pressure (PLC)
 - low steam generator water level (PLC)
 - low steam generator pressure (PLC)
 - high containment pressure (PLC)
 - high steam generator water level (PLC)
 - low reactor coolant flow (CPC)
 - manual trip (hardwired)
16. The actuation logic for the RPS interfaces with the undervoltage and shunt trip relays in the RTSS. The CEDMs receive power through the RTSS in two parallel lines from two independent, full-capacity, motorgenerator (M-G) sets connected in parallel. Each line passes through two trip circuit breakers in series (each actuated by a separate initiation circuit), so that, although both sides of the branch lines must be deenergized to release the CEAs, each side of the line can be interrupted by two separate means. The reactor trip circuit breakers trip the reactor if signals from Channel A or Channel C coincide with those from Channel B or Channel D (two-out-of-four coincidence). When the trip circuit breakers are tripped open, power is interrupted to the CEDMs, thereby releasing the

CEAs to drop into the reactor core and shut down the reactor.

An operator can trip the reactor manually by actuating two adjacent switches in the MCR to interrupt the ac power to the CEDMs. Either of two independent sets of trip pushbuttons will cause a reactor trip, as will manual reactor trip switches at the reactor trip switchgear.

17. The non-Class 1E alternate protection system (APS) provides separate and diverse reactor trip and EFW actuation logic. The APS is described in Section 7.7.

Section 7.1 describes the acceptance criteria and guidelines for the RPS. Acceptance criteria for the review of the RPS are based on the system design meeting the relevant requirements of 10 CFR Part 50 and the applicable GDC listed 10 CFR Part 50 Appendix A.

7.2.2 Conclusions

The RPS includes features that limit reactor fuel, fuel cladding, and coolant conditions to plant and fuel design limits. The system provides pre-trip alarms and trip actions that are within CESSAR-DC Chapter 15 analyses assumptions. ABB-CE has committed to ensure that instrument inaccuracies, bistable trip times, CEA travel times, and circuit breaker trip times are considered in the design of the system. The staff reviewed the ITAAC to verify that the functional requirements for system response in CESSAR-DC Chapter 15 analyses are addressed acceptably in order to confirm that these requirements are included in the I&C system design. The ITAAC review confirmed that these requirements are included, therefore, the staff finds the ITAAC to be acceptable.

The RPS conforms to the design-basis requirements of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"; the system includes the provisions to sense accident conditions and anticipated operational occurrences to initiate reactor shutdown consistent with the accident analysis in CESSAR-DC Chapter 15. Therefore, the RPS satisfies the requirements of GDC 20.

The RPS satisfies the protection system requirements for malfunctions of the reactivity control system such as accidental withdrawal of control rods. CESSAR-DC Chapter 15 addresses the capability of the system to ensure that fuel design limits are not exceeded for such events. Therefore, the staff concludes that the RPS satisfies the requirements of GDC 25.

The staff reviewed the postulated responses of the RPS to anticipated operational occurrences, and concludes that the RPS design satisfies the requirements of GDC 29.

The staff reviewed the detailed failure modes and effects analysis (FMEA) in SAR Table 7.2-5 for the various single failures considered in the design of the PPS. The FMEA was prepared assuming that one set of the redundant channels is bypassed. The staff concludes that the RPS will fail into a safe mode if conditions such as disconnection of the system, loss of energy, or postulated adverse environmental conditions are experienced. The scope of the FMEA acceptably addresses the single failure criterion of IEEE 279-1971 and the requirements of GDC 23.

The wiring in the RPS is grouped so that no single fault or failure, including either an open or shorted circuit, negates the operation of the RPS. Signals routed between redundant PPS cabinets and channels pass through fiber-optic data links. Signal conductors and power leads entering or exiting each cabinet are protected and routed separately for each channel of each system to minimize possible interactions. This design conforms to the criteria of RGs 1.75 and 1.151.

The RPS conforms to the guidelines of IEEE 384-1981, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," as supplemented by RG 1.75, "Physical Independence of Electric Systems," for the protection system independence. The RPS satisfies the requirements of IEEE 279-1971 for system independence, and satisfies the requirements of GDC 22.

ABB-CE assessed the defense-in-depth and diversity of the System 80+ I&C design, including the PPS, to address system vulnerabilities to single software errors in redundant channels and CMFs.

As stated in SECY-93-087, the staff position regarding I&C diversity for the ALWR design certification is that ABB-CE shall assess the defense-in-depth and diversity of the I&C system to demonstrate that vulnerabilities to CMFs have been adequately addressed. In performing these best-estimate assessments, ABB-CE shall analyze each postulated CMF for each event that is evaluated in the accident analysis sections of the SAR. ABB-CE shall demonstrate adequate diversity within the design for each of these events.

If a postulated CMF could disable a safety function, then a diverse means shall be provided to perform either the same function or a different function that provides adequate protection. A set of displays and controls in the MCR enables operators to manually actuate systems

performing critical safety functions and monitoring of parameters that support the safety functions.

In addressing the staff position regarding I&C diversity, ABB-CE performed a CMF analysis to evaluate the effect of failures on safety system performance. ABB-CE assumed that the CMF would simultaneously fail the reactor protection system and the ESFAS.

ABB-CE's CMF analysis included CESSAR-DC design basis events except for large- break loss-of-coolant accidents (LBLOCAs) and main steam line breaks (MSLBs) inside containment. ABB-CE excluded these events because the leak detection systems would detect flawed piping before large failures occurred. In the analysis, ABB-CE assumed nominal plant conditions at the initiation of each event, and credited best estimate responses for diverse reactor trip, EFW actuation, and for the normal control systems. ABB-CE concluded that the System 80+ plant would be brought into safe cold shutdown conditions. The CMF analysis took credit for operator actions within 15-30 minutes of the event initiation.

To monitor the status of essential safety parameters, the DIAS-P displays and processors is hardwired directly from the APCs, except for the core exit temperatures, the reactor vessel level, and the subcooling margin monitor (SMM), which are obtained from the postaccident monitoring instrumentation (PAMI) computer. Figure 7.2 depicts this configuration. The DIAS-P and the back-up ESF systems include a set of displays and controls that are independent and diverse from safety-grade computer systems.

The design includes back-up controls dedicated for manually and independently actuating ESF systems through a dedicated link that bypasses all data links, network communications, and all computers with large software applications. Switches on the main control panel (MCP) in the MCR enable operators to actuate two trains of SI and one train each of containment spray, EFW, closure of mainsteam isolation valves, closure of containment air purge valves, and closure of a letdown isolation valve. Figure 7.3 is a schematic of this configuration.

A control signal from each switch is directed to loop controllers (LCs) at the lowest level in the digital control path. The LCs are programmable logic controllers that send signals to switchgear in motor control centers and electrical distribution panels that control plant components. Under normal conditions, the LCs provide signals to plant components in response to digital signals received through the communication network. The dedicated manual signals actuate plant components by overriding the input data received from the network interface.

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This override feature is reliable because the software in the LCs typically requires less than 6 kilobytes of memory and is completely deterministic (i.e., repetitive and non-interrupt driven). ABB-CE committed to test the LCs that have the override logic to ensure that a CMF of the protection system software will not prevent the dedicated signals from actuating their associated ESF functions.

The staff reviewed this design using the criteria described in the SRM dated July 21, 1993. The staff concludes that this functional design is acceptable since ABB-CE assessed and demonstrated adequate defense-in-depth and diversity of the I&C system and provided a set of displays and controls in the MCR for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. On this basis, DSER Open Item 7.2.2.2-1 is resolved.

The staff concludes that the I&C design for System 80+ addresses the staff position on the I&C diversity issue.

The staff reviewed the CMF analysis and found that leak detection capabilities are credited in the analysis in order to exclude two accident scenarios, LBLOCAs and MSLBs inside containment. ABB-CE demonstrated that, through the use of its leak detection capabilities, the plant could be brought into cold shutdown before a large rupture. Credit for the leak detection capability in the analysis limits loss-of-coolant accidents (LOCAs) to breaks in 12-inch branch piping, and MSLBs to breaks only outside containment. Thus, diverse automatic safety injection system (SIS) actuation, which may be required in LBLOCAs and MSLBs inside containment, is not needed. This approach deviates from the staff's initial position that the analysis account for the full spectrum of breaks and locations for pipe break events identified in the accident analysis sections of CESSAR-DC.

The staff determined that credit for leak detection capability is acceptable for the CMF analysis addressing the I&C diversity issue because: (1) LBLOCAs and MSLBs inside containment in combination with a CMF in the I&C system are highly unlikely, and (2) the I&C equipment possesses sufficient diversity and simplicity including manual controls for reactor trip and actuation of emergency core cooling system (ECCS) and containment isolation in the MCR, and has a full complement of diverse instruments consistent with RG 1.97.

The staff finds the CMF analysis acceptable in addressing the events identified in the accident analysis section of CESSAR-DC because the CMF analysis demonstrates that, with credit for operator actions within 15-30 minutes of the event, the analytical results do not violate best estimate acceptance criteria of fuel performance, radiological limits

specified in related sections of CFR and SRP. Further, to enhance defense-in-depth, the staff requested that ABB-CE evaluate the LBLOCA and MSLB inside containment events to determine the time available for the operator to respond to these events. The staff reviewed ABB-CE's analysis of the operator action times assumed in the CMF analysis, and found ABB-CE's analyses to be acceptable. Therefore, the staff concludes that the CMF analysis acceptably addresses the adequacy of the I&C diversity, meets the positions for defense against common-mode failure approved by the Commission in an SRM dated July 23, 1993, in response to SECY-93-087 and the staff's proposed applicable regulations (see Section 7.1.3 above).

The RPS conforms to the guidelines for applying the single-failure criterion in IEEE 379-1977, "IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems," as augmented by RG 1.53, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems."

The systems that protect the plant are designed in accordance with ABB-CE's quality assurance program described in the Quality Assurance Program Topical Report CENPD-210 (Rev. 7), which has been found acceptable by the NRC for generic use for all ABB-CE supplied components.

The staff reviewed the software development process as described in Section 7.1.

ABB-CE will qualify the RPS as seismic Category I by performing tests and analyses to determine that the equipment, when located in the plant as proposed, will perform its functions during and after one safe-shutdown earthquake (SSE) preceded by a number of design-basis earthquakes. The tested equipment will be the same design, orientation, foundation, and importance as the installed equipment. ABB-CE committed to comply with IEEE 344-1987, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." ABB-CE's commitment is acceptable. The staff will review the use of ITAAC to verify the seismic qualification meets the requirements of IEEE 344-1987.

ABB-CE will qualify electrical equipment in accord with the program outlined in CENPD-255-A (Rev. 3), "Qualification of Combustion Engineering Class 1E Instrumentation." In September 1983, the staff reviewed and accepted this methodology, as conforming to IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and Category 1 of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." All

Figure 7.2. The System 80+ DIAS-P hardware configuration

Figure 7.3. Diverse manual ESF actuation Interface to ESF components

safety-related equipment in the RPS is qualified as Class 1E for its normal and design-basis environments.

ABB-CE's qualification guidelines for I&C equipment, NPX80-IC-QC790-00, discusses qualification of I&C equipment in mild environments. It addresses seismic, environmental, life expectancy, electromagnetic interference, surge-withstand capability, and fault isolation, and references applicable standards and RGs in the SRP, ANSI C37.90a-1974 (IEEE 472-1974), "Guide for Surge Withstand Capability (SWC) Tests," and MIL-STD-461C, 1986, "Electromagnetic Emission and Susceptibility Requirements for the Control of Electromagnetic Interference." The equipment designer can use these standards in establishing specific qualification plans and procedures. The commitments made by ABB-CE are acceptable pending verification of the design by corresponding ITAAC. The placing of NPX80-IC-QC790-00 in the public docket was designated DSER Confirmatory Item 7.2.2.4-1. ABB-CE submitted the above-referenced document in a letter to the NRC dated November 24, 1992, "Closure of System 80+ "Draft Safety Evaluation Report Issues," LD-92-115. This resolves DSER Confirmatory Item 7.2.2.4-1. Any changes to these equipment qualification commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance before being implemented. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

The RPS equipment in the control room and other control building areas (mild environments) is designed and qualified to remain functional over the range of conditions supported by the HVAC systems.

HVAC are provided for each building or room and are designed for the specific functional requirements of that individual building or room. The control building ventilation and air conditioning systems are designed to maintain the environment in the control room envelope and balance of control building within acceptable limits for the operation of unit controls, for maintenance and testing of the controls as required, and for uninterrupted safe occupancy of the control building area during post-accident shutdown. These systems are designed in accordance with the requirements of GDC 2, 4, 5, 19, and 60. Refer to Chapter 9 for the staff's discussion of the adequacy of the System 80+ HVAC.

The control building consists of the MCR, the technical support center (TSC), the computer room, the electric switchgear rooms, offices, and mechanical support equipment areas.

Instruments for the air conditioning systems provide controls and indications of the temperature and radioactivity levels of the areas sensed. Early-warning ionization-type smoke detectors are located in the supply, return, and outside air ductwork serving the control room area ventilation system.

Equipment in harsh containment environments will be qualified (as required) in accordance with the requirements of IEEE 323-1974, RG 1.89 (Rev. 1), "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," and IEEE 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The harsh environment includes temperatures from 43 °C (110 °F) to 204 °C (400 °F), a saturated/superheated steam/air mixture, radiation total integrated doses (TIDs) up to 4×10^5 Gy (4×10^7 rad) gamma and 2×10^6 Gy (2×10^8 rad) beta, and 4,400 ppm boric acid followed by a Ph of 7.0-8.5 after 4 hours using disodium phosphate. ABB-CE states that no new harsh environment equipment will be required for the ABB-CE 80+ design beyond that which has been previously qualified for the System 80 design. No digital equipment, such as remote field multiplexers (RFMs), will be located in the harsh environment. Since the portion of the sensor channels that will be subjected to the harsh environment consists of sensors and wire leads, this commitment is acceptable. An environmental qualification ITAAC will verify that ABB-CE meets this commitment.

The staff reviewed the identification of the RPS systems and components designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The staff concludes that ABB-CE has identified those systems and components consistent with the design bases for those systems. The identification of these systems satisfies this aspect of GDC 2 and 4.

The COL applicant will type test components, verify acceptable separation of sensors and channels, and confirm qualification of the cabling by the site operator to ensure the channels maintain their functional capability under applicable extremes of environment, power supplied, malfunction, and fault conditions. Sensor data shared by Class 1E and non-Class 1E systems is isolated by fiber-optic data links. The staff concludes that the loss of a single channel will not prevent the protective action of the RPS.

Sensors are connected so that blockage or failure of any one connection does not prevent protective system action. The process transducers in the containment are rated for the intended service. ABB-CE committed to qualify components that must operate during or after an accident

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for the most limiting environment for the period of time for which they must maintain their functional capability. The results of the type tests will be used to verify this requirement. This commitment satisfies the equipment qualification requirement of IEEE 279-1971 in that type test data or reasonable engineering extrapolation based on test data shall be available to verify that protection system equipment shall meet, on a continuing basis, the performance requirements determined to be necessary for achieving the system requirements and is acceptable.

The routing of Class 1E and associated cabling and sensing lines from the sensors should meet the guidelines of RGs 1.75 and 1.151. The System 80+ SAR states that the cabling and sensing lines are arranged to minimize the possibility of a CMF. This commitment requires that the cabling for the four safety channels be routed separately; however, the cables of different safety functions within the same channel may be routed together. This commitment to RG 1.75 is acceptable.

ABB-CE did not describe in the original CESSER-DC the means by which the system design meets the guidelines of RG 1.151. This was designated DSER Open Item 7.2.2.6-1. In a later submittal, ABB-CE stated that all safety-related sensing lines are protected from adverse environmental conditions. An IRWST, for example, eliminates the concern of measuring this safety-related tank level out of doors. Sensing lines for redundant channels of instruments do not share common reference legs or lines that could lead to a common cause failure of an instrumented parameter. The staff finds acceptable the means by which ABB-CE addresses RG 1.151, since the design of instrument sensing lines conforms to the guidance in RG 1.151. Since the design provides for protecting the sensing lines from adverse environmental conditions DSER Open Item 7.2.2.6-1 is resolved. Sensing lines in the station service water system (SSWS) pump structure are the responsibility of the COL applicant and, therefore, conformance to RG 1.151 for sensing lines in this structure is considered to be included in the interface requirements for the SSWS pump structure.

ABB-CE commits that low-energy signal cables will be routed separately from all power cables. Safety-related sensors will also be separated. The separation of safety-related cables requires that the cables be routed in separate cable trays. Associated circuit cabling from redundant channels is considered the same as Class 1E cabling. These commitments satisfy the criteria of IEEE 279-1971.

Cabling for redundant channels of safety-related circuits will be installed so that a single credible event cannot cause multiple channel malfunctions or interactions be-

tween channels. This commitment satisfies the criteria of IEEE 279-1971, Section 4.6, and is acceptable.

Non-Class 1E instrument circuits and cables (low level) that could be near Class 1E or associated circuits and cables are treated as associated circuits unless analyses or tests demonstrate that credible failures cannot adversely affect Class 1E circuits. The staff concurs with this commitment, and will audit the results of the analyses or tests in the performance of ITAACs.

Each channel is independent of the redundant channels. The sensors are separated, cabling is routed separately, and each channel is located in a separate cabinet, in different fire zones. The outputs from these redundant channels are to be isolated to meet the guidelines of RG 1.75 so that a single failure will not impair the system function. This commitment is acceptable.

Outputs from the redundant channels to non-safety-related areas are isolated by fiber-optic data links. Outputs from the components of the RPS to the control boards are isolated. The signals transmitted in the RPS to the DIAS, DPS, and control systems are isolated by fiber-optic data links. This commitment satisfies the isolation criteria of RG 1.75.

The staff concludes that the RPS design complies with IEEE 384-1981, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," and RG 1.75, "Physical Independence of Electric Systems."

Sensors shared by the protection systems and process control systems are isolated by fiber-optic links. The process control systems are isolated from the protection systems through the sensor cabling pathway. Portions of the protection channel used for both protection and control are classified as part of the protection system up to and including the isolation device that interfaces with the control system.

Each signal is validated to ensure that a single random failure will not affect the protective actions of the RPS, and the use of sensor signal validation logic reduces interaction between control systems and protection systems. The signal is validated in the DIAS and the DPS, which use the signals transmitted from the PPS through fiber-optic data links. The fiber-optic isolation ensures that malfunctions in the DIAS or DPS cannot affect the PPS. The signal is validated by comparing the signals from the four PPS channels with the average value of the signals. Deviations of individual signals from the average signal cause an alarm to be annunciated in the control room. The "failed" channel is automatically removed from the validation logic and tagged as "bad." The remaining signals are

then used to generate a new average, and the comparison is repeated. The staff finds that this design adequately addresses single random failures.

The staff concludes that the RPS satisfies the requirements of IEEE 279-1971 and GDC 24 for interactions between control and protection systems.

The variables for neutron flux, temperatures, and pressures in the RPS are measured directly. Level information is derived from heated junction thermocouples (HJTCs) and differential pressure measurements. Flow information is derived from measured differential pressure, reactor coolant pump (RCP) speed, and coolant temperature. These processes satisfy the requirement of IEEE 279-1971 to derive, to the extent feasible and practical, protection system inputs from signals that are direct measures of the desired variable.

RPS sensors are checked by comparison between channels. Each channel has a known relationship with the other channels of the same parameter. Cross-channel comparison is an acceptable method of checking sensor operability.

The RPS design complies with IEEE 338-1977, "Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems," and RG 1.22, "Periodic Testing of Protection System Actuator Functions." A complete channel can be tested without affecting system operability by placing the channel in bypass. Overlap in channel tests is adequately provided in the design. The sensors can be compared to similar sensors or channels that measure related information. Channel testing functions are performed continuously without lifting leads or installing jumpers.

The RPS conforms to the requirements of IEEE 279-1971 for system reliability and testability, and satisfies the requirements of GDC 21.

Individual channels may be bypassed, with a two-out-of-three logic on the remaining channels, which maintains the coincidence of two channels required for trip. This meets the single-failure criterion. The staff finds this design feature acceptable. The technical specifications (TS) will incorporate the appropriate operability requirements.

Six operating bypasses allow the operators to operate the plant without causing a RPS actuation: the DNBR/LPD trip bypass, low pressurizer-pressure bypass, high-logarithmic power level bypass, TLC DNBR CEA withdrawal prohibit (CWP) and LPD CWP bypass, the TLC RPC bypass, and the TLC CMI bypass. The operating bypasses are automatically removed when the permissive conditions are not met. The circuitry and devices which function to

remove these inhibits are designed in accordance with IEEE Std 279-1971. Therefore, the design of these operating bypasses satisfy the criteria of IEEE 279-1971 on operating bypasses.

The remote operator modules in the MCR and RPS, and by the DPS indicate test or bypass conditions and allow for the removal of any channel from service. ABB-CE's design commitment satisfies the criterion of IEEE 279-1971.

Key-locked doors control access to trip channel bypasses from the PPS cabinets. The control room operators control trip channel bypasses from the MCR. Bypassing a channel actuates an alarm to indicate the channel being bypassed. The channels being bypassed are indicated on the PPS cabinet and at the operator module. The operating bypasses have visible and audible alarms, and have features that provide a permissive range at which they can be operated. The bypassed and inoperable status indications conform to the guidelines of RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." When the permissive range is exceeded, the bypass is automatically removed. These design criteria meet the requirements of IEEE 279-1971 and are acceptable.

The setpoints for low-pressurizer pressure and low-steam-generator-pressure trips are manually reduced for the controlled reduction of pressurizer pressure and steam generator pressure during plant cooldowns. Operators initiate the setpoints from the main control board for each channel, one pushbutton for the pressurizer pressure, and one pushbutton for both steam generator pressures within one channel. This prevents the setpoint from decreasing below the actual pressure by more than a predetermined amount.

The variable low-water-level setpoint for each steam generator automatically tracks reactor power between a minimum low-power value and a maximum full-power value during plant activities. The variable setpoint is designed with a maximum ceiling and minimum floor values so that a sufficient water inventory is available after a reactor trip.

A variable-overpower trip setpoint tracks the actual reactor power between a minimum value and a high value when the reactor power changes at a slower-than-predetermined rate. The variable-overpower trip setpoint increases or decreases at a predetermined maximum rate. If the actual power increases faster than the increasing trip setpoint, the reactor will trip.

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The low-reactor-coolant-flow trip setpoint automatically tracks below the input variables by a fixed margin for all decreasing inputs with a rate less than a predetermined rate limit. The setpoint automatically increases as the input variable increases, independent of rate. The setpoint decreases at a fixed rate for all decreasing input variable changes greater than the rate limit.

This design satisfies the criteria of IEEE 279-1971 on multiple setpoints which requires that where it is necessary to change to a more restrictive setpoint to provide adequate protection for a particular mode of operating conditions, the design shall provide positive means for assuring that the more restrictive setpoint is used and that the devices used to prevent improper use of less restrictive setpoints shall be considered as a part of the protection system.

The system is designed to ensure that protective action (reactor trip) is completed once initiated. Operator action is required to clear a trip and return to operation. Protective action is initiated when the trip breakers open, and is completed when the CEAs completely insert. This design satisfies the requirement of IEEE 279-1971 for completion of protective action once it is initiated.

A manual reactor trip can be performed by depressing two sets of pushbuttons in the MCR or by remotely tripping the reactor from the RTSS, or from the RSP. This design satisfies the requirement of IEEE 279-1971 for manual initiation of protective action at the system level.

Keys or built-in features control setpoints, changes to CPC constants, calibration, and test point adjustments. Access to these items is indicated to the operator. The site operator controls access by key locks, administrative procedures, and other means to limit access. This design satisfies the criteria of IEEE 279-1971.

Indications are provided for all protective actions, including identification of channel trips. The breaker status and current indication is available to the operator. This design satisfies the criteria of IEEE 279-1971.

The operator has the means to monitor all trip system inputs, outputs, and calculations. The specific displays are provided by the DIAS and the DPS. The display modules are located in the MCR and at the RSP. This design satisfies the criteria of IEEE 279-1971.

A defective input channel is indicated by observing the system status lights or by testing. Components are repaired or replaced by placing the affected channel in bypass. This design satisfies the criteria of IEEE 279-1971.

All equipment, including panels, modules, and cables for the trip system are color coded according to channel. Physical identification enables plant personnel to recognize that PPS cabinets, RTSS, and their cabling are safety related. The labeling scheme satisfies the criteria of IEEE 279-1971.

CESSAR-DC Section 7.2.2.3.2 lists RPS equipment design criteria. In ABB-CE's original submittal, these criteria did not address reactor trip breaker tamper resistance. The staff believed that, when in service, reactor trip breaker trip attachments could be jammed without racking out the breaker and thus without causing a trip or alarm. EPRI's URD Chapter 10 Section 8.3.4.3 specifies consideration of features in the design of reactor trip breakers or their enclosures which make it difficult to tamper with the breaker in a manner that would prevent the breaker from tripping. Although conformance to the URD requirements is not mandatory, the Commission has asked for information on an applicant's reasons for deviations from the URD. The staff concluded that ABB-CE should address this URD requirement. Tamper resistance of reactor trip breaker enclosures was an open item that required resolution for the staff to complete its review (DSER Open Item 7.2.2.23-1).

By Amendment J, ABB-CE added CESSAR-DC Section 8 to Appendix 13A, "Instrumentation and Control Features for Sabotage Resistance." This section describes the sabotage resistance. The PPS includes the reactor trip breakers in rooms to which access is controlled. Within each room, cabinets that contain safety-related equipment are locked and annunciate an alarm in the control room when entered. ABB-CE stated that the reactor trip switchgear cabinets are designed such that access to the internals of the reactor trip breaker is not possible without racking out the breaker, and that a reactor trip circuit breaker cannot be racked out without causing the breaker to trip. This design precludes interference with the reactor trip function by attempts to jam the reactor trip circuit breaker. DSER Open Item 7.2.2.23-1 is resolved.

SAR Section 7.2.1.1 did not classify the RPS as a vital system. The staff considers the RPS a vital system; therefore, as required by 10 CFR 73.55(c), access to all RPS components required to trip the reactor should require passage through two barriers. Not classifying the RPS as a vital system was designated DSER Open Item 7.2.2.23-2. In a submittal dated April 30, 1992, ABB-CE amended CESSAR-DC Section 7.2.1.1 to designate the RPS as a vital system. On this basis, DSER Open Item 7.2.2.23-2 is resolved.

In Section 7.1, the staff concluded that ABB-CE adequately chose the guidelines applicable to the RPS. The staff

reviewed the RPS design and concludes that the RPS and associated systems conform to the requirement and guidelines applicable to these systems.

The staff based its conclusions on the requirements of IEEE 279-1971 and supporting standards and regulatory guidelines for the RPS. The RPS satisfies the requirement of 10 CFR 50.55a(h) for IEEE 279-1971. The staff, therefore, concludes that the RPS satisfies the requirements of GDC 24.

The design of the RPS is acceptable and meets the requirements of GDC 2, 4, 20, 21, 22, 23, 24, 25, and 29 and 10 CFR 50.55a(h).

The staff concludes that the RPS will function as required and meets the safety criteria of the applicable sections of the Code of Federal Regulations.

The staff also reviewed the PPS for quality and diversity and concludes that ABB-CE specified the appropriate quality requirements and adequate defense-in-depth and diversity for postulated CMFs. The staff determined that changes to commitments for the PPS hardware and SDP and its implementation and quality standards, and the equipment qualification program and standards would involve an unreviewed safety question and, therefore, require NRC review and acceptance before being implemented. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

The staff also concluded that performing audits in stages while implementing the ITAAC discussed in Section 7.1 is an important aspect in the final acceptance of the PPS.

7.3 Engineered Safety Features Actuation System

7.3.1 Discussion

The ESFAS actuates plant systems to terminate or mitigate plant transient conditions after a plant upset. The ESFAS consists of the electrical and mechanical components for generating the signals that actuate the ESF systems, and that send the signals to the processors and equipment in each system in the ESF-CCS.

The bistable trip functions and coincidence logic in the PPS and component control logic (CCL) in the ESF-CCS generate the required component actuation signals. The control circuitry for the components generates the neces-

sary signals to control the ESF-CCS. The ESFAS generates the following signals:

- containment isolation actuation signal (CIAS)
- containment spray actuation signal (CSAS)
- main steam isolation signal (MSIS)
- safety injection actuation signal (SIAS)
- emergency feedwater actuation signal (EFAS)

LCL in the PPS consists of full two-out-of-four coincidence logic which sends actuation signals to the ESFAS initiation relays, which connect with the ESF-CCS selective two-out-of-four logic. These actuation signals generate the PPS permissive signals for automatic ESF component operations. Each signal in the actuation logic also sets a latch when the two-out-of-four coincidence is attained to ensure that the signal is not automatically reset once it has been initiated. This feature is acceptable.

The ESF-CCS consists of four independent divisions (A, B, C, and D), except for the CIAS and the MSIS, each of which consists of two independent divisions (A and B). Each division has primary and standby processors. The primary processor controls the functions of the ESF-CCS. The standby processor tracks the primary processor. If the primary processor fails, a redundancy controller automatically transfers control to the secondary processor if the standby processor is available. This design meets the requirements for independence, redundancy, and reliability.

Each division consists of a Division Gateway and Group Controllers (A1 through An). Each segment contains redundant PLC processors, local and remote multiplexers, and the necessary communications linkages. The Division Gateway supports the master transfers switch and the operator's module interfaces, the intersystem communication data links, and a maintenance and test panel. Each Group Controller supports component control and data acquisition interfaces and related component control switches process controllers.

Local and remote multiplexing is incorporated in the ESF-CCS to reduce plant wiring requirements. Remote multiplexers are located in the MCPs, the RSP, and the remote multiplexer cabinets near plant components and instrumentation. Fiber-optic cable provides isolation where it is required to meet the channel independence criteria of IEEE 279-1971, IEEE 603-1980, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and IEEE 384-1977, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits." This design for independence by physical divisional separation and electrical isolation via fiber-optics is acceptable.

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Local and remote multiplexing can enhance data communications capabilities, although multiplexers can make the system more vulnerable to common-mode multiplexer failures. ABB-CE has adequately addressed multiplexer vulnerabilities by (1) evaluating high-quality multiplexers that meet the requirements of industrial standards, (2) assessing vulnerabilities to CMFs, and (3) ensuring adequate defense-in-depth by considering the use of hardwired essential systems controls that bypass the multiplexer networks. Section 7.1 discusses CMFs and defense-in-depth. ABB-CE will procure the safety-related multiplexers through a commercial dedication process that has been reviewed by the staff and has been found to be acceptable as discussed in Section 7.1. The evaluation of high quality multiplexers was designated DSER Open Item 7.3.1-1.

Commercial-grade computer hardware and software will be used in the safety-related systems through a process that requires:

1. requirements for supplier design control, configuration management, problem reporting, and change controls
2. review of product performance
3. receipt acceptance of the commercial-grade item from the supplier
4. final acceptance, based on equipment qualification and software validation in the integrated system

The process that has these requirements has been placed into the system ITAACs. As discussed in Section 7.1, the details of this commercial dedication process is documented in the ABB-CE document NPX80-QPS-040.1 that the staff reviewed and found acceptable. Any changes to details to the above commitments in the document would involve an unreviewed safety question and, therefore, require NRC review and acceptance before being implemented. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

The staff finds ABB-CE's commitment to implement a commercial dedication process as part of the ESF-CCS ITAAC to be acceptable, and DSER Open Item 7.3.1-1 is resolved.

The data communications networks are multidrop networks. They use active redundant cabling to maintain multiplexer interface operability should a single cable become inoperable. Communications with each ESF-CCS division are controlled by an intradivision communication network (ICN), which includes redundant cables to

maintain intersegment communications should a single cable become inoperable. Hardware failures are annunciated to minimize the time required to repair or replace the defective module. This redundancy is acceptable because it enhances hardware reliability.

For component control, each measurement channel is separated from the other three measurement channels to physically and electrically separate the signals to the ESF control logic. Cabling is separated within the cabinets, and signals to non-Class 1E systems are fiber-optically isolated. Each channel is supplied from a separate 120-V vital ac distribution bus. The independence and isolation are acceptable.

ABB-CE committed to develop system software in accordance with RG 1.152 (which endorses ANSI IEEE/ANS-7-4.3.2). The staff verified the adequacy of the software development process while reviewing the ITAAC and certain attributes of the process are contained in the ITAAC. Software issues are discussed in Section 7.1. Any changes to these commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance before being implemented. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

The ESF-CCS includes system-level selective two-out-of-four logic for ESF actuations, Group Control Logic (GCL), Component Control Logic (CCL), selective group test (SGT) logic, and diesel loading sequencer (DLS) logic.

The GCL controls groups of components. ESF functions are assigned to individual subgroup segments within each ESF-CCS division. For example, the SIS and the emergency feedwater system (EFWS) for steam generator 1 (EFW-1) are assigned to ESF-CCS subgroup segment 1, the containment spray system (CSS) and EFW-2 are assigned to subgroup segment 2. Functional assignment limits the effect of a single segment failure to ESF functions in a division. This segmentation is acceptable since segmentation limits the effect of single multiplexer or module failure to selected ESF functions in the division.

Functional assignments are further segmented within each ESF-CCS subgroup segment. For example, SIS and EFW-1 components and instruments for SIAS and EFAS-1 are assigned to separate input modules within a multiplexer to limit the effect of a single multiplexer or module failure to selected ESF functions within a division. ESF system interfaces are also confined within subgroup segments to minimize reliance on the ICN for ESF operability. For example, SIAS initiation signals and SIS component and instrument interfaces are confined to subgroup segment 1.

Failure of the ICN, therefore, would not affect SIS operation. The staff finds the independence and redundancy to be acceptable.

CCL monitors the various digital input signals (e.g., manual on-off demands, interlocks, and automatic subgroup control signals from the MCPs) and produces digital output signals to control the required components through power level interface devices. This logic also generates digital signals for status indication.

Operators perform control functions from the MCPs or the RSP. Automatic and manual component control and status indication is provided in the MCR and at the RSP. The backlighted switches and controls are linked to the ESF-CCS through remote multiplexers located in the MCPs. The MCR includes a remote ESF-CCS operators module for backup in the event of a switch or multiplexer failure. This panel enables the operator to control components by selecting items from a menu on a qualified video display unit. This design satisfies the requirement of IEEE 279-1971 to provide information pertinent to the status of protection system components. Section 7.1 discusses issues concerning the implementation of the software.

The RSP can be used to control the plant to achieve hot-standby conditions when the MCR cannot be used. The RSP control devices are linked to the ESF-CCS through multiplexers in the RSP. A remote ESF-CCS operators module can serve as a backup if a control switch or multiplexer fails. This remote operator's module can also control all ESF-CCS components necessary to achieve cold shutdown. Fiber-optic cable is used to prevent a fault from propagating to the ESF-CCS from the MCR or the RSP. This design provides the necessary isolation and is acceptable.

As a defense-in-depth measure, local control switches (LCS) are independent from the ESF-CCS for components essential to hot shutdown. Only manual control (i.e., ON/OFF, START/STOP, OPEN/CLOSE) is provided through these switches. The LCSs are field wired for direct control of components or motor-control-center component actuators, and are located near the actuated components in such field locations as the motor control centers. The LCS may also be used for test and maintenance. This design satisfies IEEE 279-1971. Additional diversity requirements are described in Section 7.1.

Process controllers are supported where required from ESF-CCS subgroup segments to enable the operator to control continuous process control functions (such as valve modulation and auto/manual mode selection). This design satisfies the criteria of IEEE 279-1971 for manual initiation.

Each ESF-CCS equipment cabinet includes a maintenance and test panel. The operator observes this panel for ESF-CCS equipment status and uses it for ESF-CCS maintenance, testing, and diagnosis. Each panel includes a master transfer switch (MTS) for transferring control from the MCR to the RSP. Each component is preprogrammed to remain in its current state or go to a predetermined safe state when the MTS is actuated. The staff accepted ABB-CE's commitment in the DSER to install redundant MTSs near the MCR exits and at the RSP. This was a confirmatory item. The SAR did not include sufficient design information for the staff to determine the adequacy of the MTS design. The lack of detailed information of the MTSs was designated DSER Open Item 7.3.1-2. ABB-CE submitted a more detailed design description in response to this open item. The staff reviewed the MTS design as discussed in Section 7.4.6.

Setpoint values are administratively controlled and automatically monitored continuously. The fixed setpoints are adjusted at the PPS cabinet. Access for setpoint adjustment is limited by keylock with the access annunciated by the DIAS. The setpoints may be displayed at the PPS cabinet and through the DPS cathode ray tube (CRT) displays in the MCR.

The components of each ESFAS are segregated into functional groups, such as a valve group and a pump group. The operator in the MCR performs ESFAS selective group testing. This testing overlaps the PPS automatic testing of the ESF-CCS selective two-out-of-four coincidence logic and includes complete testing of the ESFAS through to the actuation of the group components. Testing is conducted one group at a time, thus preventing the undesired actuation of an entire ESF system during testing. Since this testing causes components in a group to actuate, an ESFAS signal from the PPS will not be impeded and the ESF system will be fully actuated regardless of the testing process. The ESF system is designed to be tested without lifting leads or inserting jumpers. This testing capability satisfies the criteria of IEEE 279-1971 that require that capability shall be provided for testing and calibrating channels and the devices.

7.3.2 Component Actuators

The ESF-CCS controls five types of components:

- solenoid-operated valves
- motor-operated valves
- contactor-operated components
- circuit breaker-operated components
- modulating components

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The ESF-CCS executes the control logic necessary to energize solenoids upon detecting the appropriate state (open or closed) to which the solenoid corresponds. The solenoid operator receives a position status signal for status indication and interlocking with other components. The control logic senses the state of the output relay and the continuity monitoring circuit for the output. A digital output module in the RFM sends the relay signal to energize the solenoid.

Modulating valves with solenoid operators have electropneumatic modulators to allow continuous valve positioning. This operator uses three types of signals: a solenoid-energized signal, an analog position signal, and a control signal.

Motor-operated valves receive position status, control signals, and a signal to determine when the opening and closing coils are deenergized.

Contactors-operated components monitor position status, a control signal, output status, and continuity to indicate contactor status, contactor discrepancies, and inoperable components for the operators. The component inoperable signal prevents latches from being reset in the control logic, and informs the operator of the inoperable status.

Circuit breakers control most loads requiring voltage greater than 480-V ac. Electrical fault protection interfaces and rack-out or test position interlocks are wired outside the ESF-CCS. Components operated by circuit breaker receive a position status signal linked to a digital input module in the RFM, a control signal indicating the state of the output relays and the continuity circuits for each output, a signal from the overcurrent relay to provide fault indication, and a signal from control power monitoring contacts for closing and trip circuits. These signals are logically combined to indicate circuit breaker status, discrepancies, and inoperable components.

Modulating components include electropneumatically-actuated and electrohydraulically-actuated components (pumps or valves) that require only analog signal inputs for continuous control. Pilot solenoids do not send discrete state controls. Valve status signals report whether the valve is fully opened or fully closed, and the analog position of the valve, where required. Pump signals relay the on/off status and, where applicable, the turbine speed of the pump driver. The signals connect with the RFM for the component.

Actuation signals generated by the selective two-out-of-four logic in the ESF-CCS actuate groups of ESF system components required by the ESFAS function. These components include solenoid-operated valves, motor-

operated valves, and the motors of pumps. Valves and pumps for a particular ESFAS function are grouped within an ESF-CCS division in such a way that the required component groups are actuated by the appropriate logic. The ESFAS is linked within the CCL for each component. The application of thermal overload protection devices in Class 1E motor-operated valve circuits is in compliance with RG 1.106 in that these devices are not used in safety-related motor-operated valve circuits and are used to provide alarm functions.

7.3.3 Diesel Loading Sequencer

The diesel generators (DG) are the emergency backup source of power. These generators must be sequentially loaded to meet large power demands. The emergency equipment and systems are loaded in groups to avoid instantaneous overloading of the DGs. The equipment is energized as quickly as possible to minimize overall plant disturbances.

The diesel loading sequencer (DLS) is designed assuming plant accidents occur before, during, or any time after a loss of offsite power (LOOP) or a blackout. The ESF equipment required for a design-basis accident (DBA) is energized within a predetermined time after an accident has occurred to maintain the plant within its design limits. The equipment required to be loaded depends upon the accident scenario. Several load groups of equipment may be required if multiple ESF systems are needed to mitigate an accident.

The load sequencing is automatically varied to accommodate different loading requirements. If, during a load sequencing operation, an ESF system is actuated, the load sequencer interrupts the loading of a nonessential load and starts loading the appropriate ESF load group. If an accident does not occur during the load sequencing, the nonessential loads are loaded without unnecessary delays.

The load sequencer is fully testable during online plant operations.

The load sequencer initiation logic monitors various plant electrical buses to determine abnormal power conditions. Upon detecting a condition that indicates a requirement for the emergency DG, a loss-of-power signal sets all sequencer output latches. This sends load shed trip signals to the necessary plant equipment. A DG automatic start signal is also transmitted to the DG upon loss of power, SIAS, CIAS, or EFAS. When the DG is ready to accept the first load group, a DG circuit breaker close signal is transmitted to connect the DG to the plant bus. After the circuit breaker is closed successfully, the equipment loading sequence begins.

The basis of the loading sequence logic is an eight-step counter. Additional steps are added, as necessary, to control the sequencing for all nonaccident equipment. When the DG breaker is closed, the counter advances, one step at a time, at equal intervals. The interval is a clock pulse adjusted for discrete digital increments. At each step of the counter, one sequencer output latch is reset, thereby removing the load shed signal from that load group and reenergizing the required plant equipment. During nonaccident plant conditions, the counter will advance, uninterrupted, to reset the eight output latches, thereby reenergizing the eight nonaccident load groups.

Three priority load groups handle plant ESF equipment, with additional load groups added, as necessary. The priority interrupt logic continuously monitors ESF actuation signals from the PPS. If an ESF is actuated, the clock pulse from the load sequence logic is redirected from the step counter to reset the sequencer output latch for the appropriate priority load group. The required ESF load group is energized, interrupting the nonaccident loading sequence. Once a Class 1E division load group is energized, that group is not shed upon the occurrence of an accident.

The nonaccident loading sequence may be interrupted if multiple ESF actuations occur at different times. A priority between ESF load groups is established so that if two or three ESF actuations occur during any one clock cycle, the ESF load groups will be sequenced in successive steps and in the established priority order. After the ESF load groups are loaded, the nonaccident sequence resumes.

ABB-CE's initial description of the DLS did not state whether the DLS is based on a relay, a PLC, or a mini-computer system. ABB-CE did not give sufficient details on the DLS location; inputs such as for loss of voltage, degraded bus voltage, or A/D conversion; and number of DLSs. Consequently, the staff could not verify whether the DLS hardware and software is implemented acceptably. This was designated as DSER Open Item 7.3.3-1.

ABB-CE subsequently stated that the DLS is a PLC-based system. The inputs from the division buses are received in discrete or digital form. The DLS monitors voltage from the division 4.16 kilovolt bus, ESF signals, EDG voltage and frequency, the status of EDG breakers, and the status of bus load breakers. The undervoltage relays monitor the buses for an undervoltage or a degraded voltage condition, which could result from a loss or degradation of offsite power or onsite power. When an undervoltage condition occurs, the DLS initiates an automatic start of the EDGs and initiates load shed signals to all large loads. These loads are described in Section 7.3.1.1.2.3(A), (B), and Section 8.3.1.1.4.6 of the

CESSAR-DC. Based on the information provided in the SAR, DSER Open Item 7.3.3-1 is resolved.

The inputs used by the DLS from the PPS are the ESFAS: SIAS, CSAS, and EFAS 1 and 2. The ESFAS inputs are used in the priority interrupt logic to sequence accident-related loads, and to initiate an automatic start of the EDG. When an ESFAS occurs with no undervoltage condition, the DG is started as a standby source of power. Since no power is lost, the normal (nonaccident) loads are not shed and; therefore, only the accident loads need to be sequenced onto the bus. Because bus power was not interrupted, sequencing of these loads begins immediately, before the DG attains normal operating parameters.

The software and hardware implementation of the DLS is within the scope of the ESF-CCS ITAAC discussed in Section 7.3.1. The staff reviewed ABB-CE's computer system development process as discussed in Section 7.1. Any changes to these design features would involve an unreviewed safety question and, therefore, require NRC review and acceptance before being implemented. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

7.3.4 Bypasses

The System 80+ design has two types of bypasses: bistable trip channel bypasses and operating bypasses.

The PPS includes a bistable trip channel bypass for maintenance and testing.

The operating bypass for low pressurizer pressure permits an orderly startup and shutdown of the plant and allows system tests at low pressure. This manually actuated bypass cannot be initiated if the pressurizer pressure exceeds 400 psia, and is automatically removed if the pressure exceeds 500 psia. This bypass disables the low pressurizer pressure trip from SIAS and CIAS.

Auxiliary and supporting systems for the safety-related I&C system cause a system-level bypass indication when these systems are bypassed or deliberately made inoperable. A bypass indication is provided for the safety-related system affected by the bypass. This indication of bypassed systems satisfies the IEEE 279-1971 requirement of continuous indication in the control room if the protective action of some part of the system has been bypassed or deliberately rendered inoperative for any purpose.

The bistable trip channel interlocks for ESFAS, located in the PPS, prevent the operator from bypassing more than one trip channel or a trip parameter at a time. Different

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trip parameters may be bypassed simultaneously, either in the same channel, or in different channels. If a complete channel is disabled such as loss of a vital power supply, the operator will manually bypass all trip bistables for that channel, thereby rendering that channel bypassed. This is a manual action from the operator's control panel.

The operators receive the bypass status on displays in the control room, and through displays on the DPS. This satisfies the guidance in RG 1.47 as an acceptable method for implementing the requirements of Section 4.13 of IEEE Std 279-1971 and Criterion XIV of Appendix B to 10 CFR Part 50 with respect to indicating the bypass or inoperable status of portions of the protection system, and auxiliary or supporting systems that may be operable for the protection system or the system it actuates to perform their safety-related functions.

During PPS testing, interlocks prevent disabling of more than one redundant protection function at a time and prevent maintenance personnel from inadvertently causing unwarranted ESFAS signals. The bypasses satisfy IEEE 279-1971.

7.3.5 Redundancy

Circuits in the four PPS channels initiate four independent channels for each parameter from the process sensor. Four redundant ESF-CCS divisions operate four (or two) completely redundant ESF trains. Redundant components of the engineered safety system are assigned to redundant ESF-CCS divisions to maintain the desired level of design redundancy.

Redundant flow paths in the system design ensure cooling capability under single-failure conditions. These flow paths are assigned to different divisions of the ESF-CCS to maintain flow availability under single-failure conditions within an ESF-CCS division.

A redundant flow path may contain two valves in series to preclude spurious flow initiation upon single failures. Each valve is assigned to an independent ESF-CCS division so that a single failure within a division will not cause spurious flow. Spurious flow is prevented while independent redundant ESF flow paths are maintained by assigning selected components in ESF system trains A and B to ESF-CCS divisions C and D, respectively. The redundancy of the systems satisfies IEEE 279-1971.

7.3.6 Diversity

Section 7.2 presents the staff's position on diversity in systems that depend on software for safety functions.

ABB-CE performed a CMF analysis of the ESF-CCS to verify sufficient defense-in-depth and diversity in the design for vulnerability to common-mode and software errors. The staff evaluated the adequacy and results of the CMF analysis, as discussed in Sections 7.1 and 7.2.

The non-safety-related APS gives the operator diverse capability to actuate and control the EFWS through the APS alternate feedwater actuation system (AFAS). The APS operates with the ESF-CCS by sending isolated AFAS signals to the CCL of the EFWSs. Section 7.7 discusses the APS.

The non-safety-related process-CCS includes diverse equipment for controlling critical plant parameters and is available through the video display units in the MCR and the RPS, and at the LCSs throughout the plant near the equipment actuators. The process-CCS is controlled with diverse software and hardware equipment. Section 7.7 discusses the process-CCS.

7.3.7 Testing

The design provides for periodic testing of the ESFAS to cover the trip actions from the sensor through the actuated device. The system testing does not interfere with the operation or protective functions of the ESF-CCS. The staff reviewed the sensors through the actuated equipment, including the capabilities for LCL tests, initiation logic tests, actuating logic tests, SGTs, bypass tests, and response time tests. The staff's acceptance of system testing is subject to the staff's acceptance of the TS. This was designated DSER Open Item 7.3.7-1. Staff acceptance of the TS is discussed in Chapter 16.

During reactor operation, the DIAS and the DPS continuously monitor the sensors to detect sensor failures or degraded sensor signals. Section 7.2 includes additional details regarding sensor signal validation and bistable trip testing. The DPS performs additional testing of the bistable functions for the process control setpoints and alarms.

The DPS continuously monitors setpoints and activates alarms upon finding excessive setpoint deviation between redundant channels. While testing selected groups, operators periodically verify PPS bistable trip accuracy and interlock performance by manually varying the digital interlocking parameter from the operators module in the MCR. Analog-to-digital conversion accuracy is also periodically verified at the operator's module during sensor testing. The overlap of testing enables operators to verify PPS bistable trip accuracy and interlock performance.

The PPS test function automatically tests the initiation logic for each engineered safety system to determine its ability to generate an initiation signal. Testing begins by interrogating the status of the input signals to the logic and the state of the output. The test function compares the value of the output signal with the value of the input signal. Discrepancies are annunciated and a message is provided to the operator to describe the error. If there is no discrepancy, the testing function continues. Based upon the known inputs, the testing function will generate combinations of input signals and monitor the outputs of the logic for correctness. The testing function cannot change a genuine coincidence signal.

Testing is performed one channel at a time to avoid inadvertently actuating equipment with the testing function. This testing is done with the ESFAS initiation relay testing.

The ESF-CCS actuation logic is a selective two-out-of-four circuit controlled by signals from the initiation relays from the four PPS channels. These channels may be tested automatically. The testing process is similar to that described above. Before the automatic test function applies any signals to the system, the logic determines the status of the initiation circuit outputs then determines the expected output. If there is a discrepancy, the logic actuates an annunciator, and sends more detailed information to the operator upon demand. If the outputs are correct, the testing system sends an initiation signal to the ESF actuation train(s). The test system then monitors the feedback signals from the trains to confirm proper operation. If a fault is detected, it is annunciated, and a message is sent on demand.

The initiation relay test is only performed in one PPS channel at a time. An interlock in the PPS channels ensures that only one channel at a time can be tested. If a channel has tripped, the testing function will not test an opposite channel, which would cause an inadvertent actuation.

A delay circuit in each channel eliminates spurious signals. A channel will not be actuated unless the actuation signal lasts for a period longer than the delay. This delay enables the ESF-CCS actuation logic to receive brief test signals from the PPS, process the signals, and return them to the PPS for detection of an initiation signal failure or the loss of an actuation signal to a subgroup of equipment. The PPS sequentially generates short-duration signals for each ESFAS actuation signal (A, B, C, then D). A channel trip requires simultaneous actuation of (A or C) and (B or D); consequently, the testing logic cannot cause an inadvertent actuation of an ESFAS channel.

The PPS processes the returned signal to detect both the presence of a valid actuation signal and the absence of invalid actuation signals. Invalid responses are detected as they occur. When an actuation channel is manually actuated at the ESF-CCS, a discrepancy between the PPS initiation signals and the state of the actuation channel is automatically detected.

The operator in the MCR manually tests selected ESFAS groups. This testing overlaps the PPS automatic testing of the ESF-CCS selective two-out-of-four logic and includes complete testing of the ESFAS through to the actuation of the components in the selected group. The operator tests one group of ESF components at a time, thereby preventing a complete (and undesired) actuation of an ESF system. Since this testing causes ESF components to actuate, an ESFAS signal from the PPS will not be impeded, and the ESF will proceed to full actuation.

The diesel load sequencer may be tested while the DLS is on line. During normal operation, all output control signals are disabled, allowing all logic functions to be tested without disturbing plant equipment. The outputs become enabled automatically when a valid initiation signal is received. Consequently, testing may be conducted without impeding load sequencer operation in the event of a station black-out condition. There are three distinct test phases: automatic testing, input testing, and load shed testing.

ABB-CE's testing processes satisfy the requirements of IEEE 279-1971 for providing the capability for testing and calibrating channels and devices used to derive the final system output from the various channel signals.

7.3.8 Vital Instrument Power Supply

Chapter 8 discusses the vital instrument power supply design.

7.3.9 Actuated Systems

The ESF systems are maintained in a standby mode during normal plant operations. The ESFAS generates actuation signals to ensure that the ESF-CCS performs the required protective actions. In CESSAR-DC Section 7.3.1.1.10, ABB-CE briefly described the I&Cs of each ESF system. The staff reviewed the ESF system actuations for the design-basis events and the monitored parameters required for ESF systems actuations.

7.3.9.1 Containment Isolation System

The containment isolation system enables the CIAS to isolate fluid systems that pass through containment penetra-

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tions so that any radioactivity released into the containment after a DBA will be confined within the containment. A high containment pressure signal or a low pressurizer pressure signal sends a CIAS and sends a SIAS.

Valves that must be isolated following a CIAS are installed with either air-operated controllers or motor-operated controllers. Air-operated valves fail to the position of greater safety upon loss of instrument air. Motor-operated valves are powered from the emergency power sources. Valves are located to protect against missiles and environmental concerns, such as radiation and pipe whip. Dual valves, with one valve inside the containment and one valve outside the containment, is used for all isolating valves. Lines that must remain in service following an accident have at least one remote manual valve.

7.3.9.2 Containment Spray System

The CSS is a safety-grade system designed to reduce containment pressure and temperature caused by a MSLB inside the containment or LOCA inside the containment and to remove iodine from the containment atmosphere following a LOCA. Iodine removal is required so that, in the event of containment leakage, activity at the site boundary will be reduced.

The IRWST supplies spray water to the CSS, which consists of two independent trains (one containment spray pump, one containment spray heat exchanger, one independent spray header, and associated valves and instrumentation per train).

The containment spray pumps start upon receipt of a SIAS or CSAS, which is actuated by high containment pressure or low pressurizer pressure. The pumps discharge through the containment spray heat exchangers and the spray header isolation valves to their respective spray nozzle headers located in the upper regions of the containment. Spray flow to the spray headers is not provided until a CSAS automatically opens the containment spray header isolation valves. The CSAS is actuated by a high-high containment pressure signal.

The CSS can receive power from the emergency power source (the emergency DGs), the plant turbine generator (the onsite source), and the plant startup power source (offsite power). Power connections are through a minimum of two independent buses. Each CSS train is powered from an independent bus.

The independent trains A and B are physically separate to ensure a single failure will not disable both trains. This includes the power systems for the I&C systems. Associated circuits from redundant channels are either separated,

provided with isolation devices, or shown to be not affected by a single failure. Routing accounts for pipe whip and postulated ruptures.

The environmental control system for each independent train is powered by the emergency power source for that train through power a minimum of two independent buses. The CSS instruments have two independent vital instrument power sources.

Provisions are made for the detection, containment, and isolation of the maximum expected leakage from a moderate energy pipe rupture in one train. The process instruments in the control room assist in the operator controlling conditions after a LOCA.

The CSS I&C equipment is environmentally qualified for LOCA conditions in the containment. These conditions are 177 °C (350 °F) for 10 minutes, 149 °C (300 °F) for the following 16 hours, 520 kPa (60 psig) for 16 hours, a saturated and superheated mixture of steam and air, 3.5×10^5 Gy (3.5×10^7 rad) gamma and 2.0×10^6 Gy (2.0×10^8 rad) beta TID over 60 years (plus LOCA), and 4400 ppm boric acid spray with a pH 7.0-8.5 after 4 hours using disodium phosphate. These parameters are acceptable for environmental qualification in accord with the requirements of 10 CFR 50.49 as discussed in Section 3.11.

7.3.9.3 Main Steam Isolation System

The main steam isolation system will isolate the steamline piping and the main feedwater piping in a steam generator after a steam generator tube rupture, a main steamline break, or a main feedwater system upset. Initiation of a MSIS closes all valves in the associated main steamline piping up to and including the main steam isolation valve (MSIV), except for the main steamline safety valves, the atmospheric dump valves (ADVs), the EFW steam turbine admission valves, and the EFW control valves. An MSIS is initiated upon receipt of a high-containment-pressure signal, a low-steam-generator-pressure signal, or a high-steam-generator-water-level signal.

7.3.9.3.1 Main Steam Isolation Valves

The MSIVs will isolate the main steam lines outside the containment after a main steamline break. The MSIV in each main steam line is remotely operated and can remain tightly closed under the main steamline pressure, temperature, and flow resulting from the transient conditions of a pipe break in either direction of the valves. The MSIV will fail closed on loss of power and loss of instrument air, and will automatically close upon receiving an MSIS. There are four MSIVs, one in each of the two steam lines for each steam generator.

The bypass valve around one of the MSIVs on each steam generator allows the main steam lines to be warmed downstream of the MSIVs before opening the MSIVs. The bypass valves also close automatically upon receipt of an MSIS. The MSIV bypass valve control circuits are designed so that no single electrical failure results in a spurious opening of the valve. No single failure of the control circuits will prevent the MSIV bypass valves from closing.

The MSIVs are controlled by a separate system independent of the protection system. An operator control is provided locally, in the MCR, and at the RSP. This control allows the operator to manually open and close the MSIVs and test the MSIV operation (from the MCR only). The control includes open/close indication lights.

Each MSIV has two physically separate and electrically independent closure solenoids to provide a redundant means of valve operation. The MSIS is sent to each solenoid. An electrical or mechanical malfunction of one solenoid will not prevent the MSIV from closing.

The available air supply for the valve pneumatic operators is clean, dry, oil-free, and delivered at the point of use under system full-flow conditions at a pressure of 580 kPa (70 psig) to 825 kPa (105 psig). The pneumatic lines and fittings are designed for a pressure of 1140 kPa (150 psig). An air reservoir for each MSIV prevents accidental closure if the normal air supply inadvertently fails.

During initial startup and periods of unit shutdown, the tripping mechanisms for the MSIVs will be tested for proper operation in accordance with the TS. The valves will be periodically tested for leakage and freedom of movement during plant operation in accordance with ASME Section XI.

7.3.9.3.2 Main Feedwater Isolation Valves

The redundant main feedwater isolation valves (MFIVs) meet the single-failure criteria. These valves isolate all steam generator interconnecting lines to preclude coolant from blowing down the lines to both steam generators after a pipe rupture in one steam generator. The MFIVs will stop all forward feedwater flow within 5 seconds after receipt of an MSIS, even if a single failure occurs. The MFIVs are capable of being in-service tested. The MFIVs are remotely operated and capable of maintaining a leak rate of less than 1000 cc/hr (0.004 gpm) under the main feedwater line pressure, temperature and flow conditions of a postulated pipe break in either direction of the valves. Each MFIV actuator is physically and electrically independent of the other actuator so that failure of one actuator will not cause the other to fail.

7.3.9.4 Safety Injection System

The SIS provides an emergency source of coolant to the RCS if normal coolant inventory is lost. The actuation system consists of redundant trains A, B, C, and D and will start the SIS upon receiving an SIAS indicating low RCS pressure or high containment pressure. The SI will begin automatically if the SIS receives appropriate signals from four independent pressurizer pressure sensors and four independent containment pressure sensors. The SIS uses a local two-out-of-four coincidence logic for each parameter. An SIAS may also be manually initiated from the control room. During startup and shutdown operations, the pressurizer low-pressure setpoint may vary to allow the system pressure to be decreased without actuating the SIS.

The I&C system for each train is physically and electrically separate and independent. The SIS can sustain the loss of an entire train and still provide the required protective actions. The SIS I&C is designed to operate under all plant conditions. The SIS can be remotely controlled from the MCR, the RSP, and the local control stations.

The SIS consists of redundant active and passive coolant injection systems. The active portion of the SIS consists of four mechanically separated trains, each consisting of a SI pump and associated valves. Each pump has its own connecting line to the IRWST, and its own discharge line to a direct vessel injection nozzle on the reactor vessel. The passive portion of the SIS consists of four identical pressurized SITs.

The SIS will also inject highly borated water into the reactor vessel to mitigate accidents other than LOCAs. SI will be initiated in the event of a steam generator tube rupture, steamline break, or a CEA ejection event. The borated water injected by the SIS provides inventory and reactivity control for these events. The SIS is also an alternate means of decay heat removal for those events that are beyond the licensing design basis in which the steam generators are not available (loss of normal heat sink). The SIS and the safety depressurization system (SDS) (Section 7.4) perform feed-and-bleed cooling of the RCS.

The electric power required for the SIS controls and instruments is supplied through two Class 1E ac buses. The EDGs are an alternate source of power.

Where required to ensure proper system alignment, the open or closed position of remotely actuated valves is administratively controlled in the control room. The control room operators control the status of all valves (except the check valves) in the SI flow path that are not required to operate upon initiation of the SIS to ensure

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these valves are maintained in the appropriate SI position during plant operation. The open or closed position of the SIT isolation valves is indicated in the MCR.

7.3.9.5 Emergency Feedwater System

The EFWS supplies emergency cooling water to the steam generators to ensure the capability to remove decay heat from the RCS after a loss of main feedwater or other event that requires the removal of heat from the primary coolant system through the steam generators. This includes the loss of offsite ac power.

The EFWS mechanical components are configured into two separate trains aligned to feed their respective steam generators. Each train consists of a 100-percent-capacity motor-driven pump subtrain, a 100-percent-capacity steam turbine-driven pump subtrain, an EFW storage tank, associated instrumentation, controls, piping and valves, and a cavitating venturi.

Each EFW pump is capable of supplying feedwater to both steam generators during plant cooldowns to shutdown conditions. While the EFW trains are normally dedicated to each steam generator, EFW pump discharge crossover piping allows feeding both steam generators from a single train. The crossover piping isolation valves are normally locked closed, but may be opened manually.

The steam-driven EFW pump speed can be controlled from the MCR, the RSP, and manually at the governor. The turbine may be stopped by tripping the trip and throttle (stop) valve from the MCR or the RSP. The turbine is tripped automatically for turbine overspeed protection by an electrical trip at 115 percent of rated speed, and by a mechanical trip at 125 percent of rated speed. The electrical trip may be reset from the control room or the remote shutdown station. The mechanical trip can be reset only at the valve.

The instruments, controls, and valve operators essential to the operation of the EFWS are powered by battery-backed Class 1E power. The batteries are capable of supplying power to the steam-driven EFWS pump subtrains for 8 hours. The motor-driven pump subtrains can be powered using the alternate ac (AAC) source of standby power for extended station blackout periods.

The EFWS actuation logic and Class 1E power sources are composed of redundant trains A, B, C, and D. The I&Cs of each train are physically and electrically separate and independent. The EFWS can sustain the loss of an entire train and still provide the required protective actions. The EFWS I&Cs are designed to operate under all plant conditions.

The EFWS is initiated either manually, automatically by an EFAS from either steam generator (EFAS-1 for steam generator 1, EFAS-2 for steam generator 2), or by the APS AFAS. The APS is discussed in Section 7.7. An EFAS is generated by a low-steam-generator-level signal to actuate EFWS or a high-high-steam-generator-water-level signal to close the feed valves. The EFWS can be controlled from either the control room or the RSP. Some portions of the EFWSs also can be manually controlled from local stations at the individual components. The EFWS uses a four-channel control scheme to preclude inadvertent actuation in the event of a single failure. The four-channel design includes the initiation logic, actuation logic, and the power supplies.

On a low steam generator level, the EFAS signal for that steam generator starts the dedicated motor-driven and steam-driven EFW pumps and opens the EFW isolation valves and flow control valves, causing the system to operate at full-flow. EFW does not start to flow into the steam generator until the steam generator level reaches its low-level setpoint. The actuation signal that opens the valves clears automatically when the steam generator level reaches its normal operation setpoint. When the actuation signal clears, the valves remain in their open position; however, the operator can manually control the flow rate. Otherwise, full flow continues. If the steam generator level again decreases to the low-level setpoint, the EFAS signal reactuates, causing resumption of full-flow system actuation.

To prevent the steam generator from being overfilled, the ESF-CCS includes a high-steam-generator-level interlock to automatically close the isolation valves. This interlock is active only when the EFAS actuation is not active (i.e., before the steam generator level decreases to the low-level actuation setpoint, or after the steam generator level has been recovered to its normal operating setpoint). This interlock also protects against steam generator overfill caused by erroneous operation of the EFWS by the operator or the APS.

The EFWS piping and associated structural supports and restraints will be designed so that a ruptured EFW line or a closed isolation valve will not reduce the capability of the ESFAS or PPS.

Non-safety-related portions of the I&C system are designed so that any failure will not degrade of any safety-related equipment function. The EFW flow rate, pump discharge pressure, pump suction pressure, and pump turbine inlet pressure are displayed in the MCR and at the RSP.

The EFW design satisfies the requirements of IEEE Standard 279-1971 (10 CFR 50.55a(h)), and the requirements of 10 CFR 50.34(f)(2)(xii).

7.3.9.6 Containment Hydrogen Recombiner System

The containment hydrogen recombiner system (CHRS) consists of two independent, full-capacity, parallel loops that include suction points within the containment to eliminate stagnant pockets of air where hydrogen could accumulate. Recombiner inlet connections in the IRWST remove hydrogen produced by radiolysis in the IRWST. The hydrogen concentration sensors satisfy the Category 1 guidelines of RG 1.97, and the requirements of TMI Action Item II.F.1, as required by 10 CFR 50.34(f)(2)(xvii)(c).

In the event of a LOOP, the CHRS will receive Class 1E power from the EDGs.

Access and shielding are provided to the areas where the portable hydrogen recombiner and control panel skids are to be placed and areas where coupling operations are required. The CHRS design is discussed in Section 6.2.

Redundant hydrogen analyzers monitor the performance of the recombiners by measuring the hydrogen concentration of the incoming gas from the containment and the recombiner discharge. The hydrogen analyzers are independent of the hydrogen recombiners and are permanently installed to monitor hydrogen concentration throughout the accident. Hydrogen monitors are discussed in Section 7.5.

The CHRS prevents the concentration of hydrogen in the containment from reaching flammability limits after a design-basis LOCA. The CHRS is designed to be manually initiated upon indication of high hydrogen concentration. The DPS monitors hydrogen concentration and sends the information to the operators in the MCR and at the RSP for viewing on the DPS CRT displays. ABB-CE did not state in the DSER whether hydrogen concentration is provided as a Class 1E alarm. The staff initially determined that this system should be Class 1E as required in IEEE 279-1971 (DSER Open Item 7.3.9.6-1). Emergency procedures require the hydrogen recombiners to be connected and activated within 84 hours of a LOCA, which is before reaching the high hydrogen concentration setpoint for design basis events. The hydrogen recombiners then operate continuously. Since the operator is not required to operate the hydrogen recombiners after they are actuated, (e.g., upon detecting a high hydrogen concentration) no operator action is required to perform this safety function after the recombiners have been

actuated. Consequently, the staff concludes that the alarm on high containment hydrogen is not required to be a Class 1E alarm. This closes DSER Open Item 7.3.9.6-1.

7.3.10 Conclusions

At the time that the DSER was completed, the staff lacked sufficient design information to conclude that the ESFAS and ESF-CCS met the relevant requirements of GDC 2, 4, 20 through 24, 34, 35, 38, and 41 and IEEE 279-1971 (as required by 10 CFR 50.55(h)). The staff noted that a significant portion of the system design is software-based, and could not be reviewed. Subsequently, the staff evaluated software aspects of the system design as part of ABB-CE's ITAAC submittal. The staff verified that other portions of the ESFAS and ESF-CCS design meet the relevant criteria based on the following.

The staff reviewed the ESFAS and ESF-CCS for conformance to the applicable regulatory guidelines in Section 7 of NUREG-0800 (SRP) and found reasonable assurance that the systems will conform to the applicable guidelines.

The staff also determined which systems and components of the ESFAS and ESF-CCS are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. ABB-CE designed these systems consistent with the design bases for these systems. Sections 3.10 and 3.11 describe the qualification programs to demonstrate the capability of these systems and components to survive these events. ABB-CE chose these systems and components in a manner that satisfies this aspect of GDC 2 and 4.

The ESFAS design commitments conform to the design-bases requirements of IEEE 279-1971, and the systems include provisions to sense accident conditions and anticipated operational occurrences, and will cause the appropriate ESF-CCS and essential supporting systems to operate consistent with the accident analyses in CESSAR-DC Chapter 15. Therefore, the ESFAS satisfies the requirements of GDC 20.

The ESFAS conforms to the guidelines for periodic testing in RC 1.22 and IEEE 338-1977 (as supplemented by RG 1.118). The bypassed and inoperable status indications conform to the guidelines of RG 1.47. The ESFAS conforms to the guidelines and application of the single-failure criterion in IEEE 379-1972 (as supplemented by RG 1.53). In performing the FMEA, ABB-CE considered the removal from service of any component or channel. Therefore, the staff concludes that the system satisfies the testability and the single-failure criteria of IEEE 279-1971

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and GDC 21. Consequently, the issue of GDC 21 in the areas of high functional reliability of the system and single failures in software, which was designated DSER Open Item 7.3.10-1, is resolved.

The staff concludes that the ESFAS design conforms to the guidelines in IEEE 384-1981 (as supplemented by RG 1.75) for the protection system independence. The design commitments for the digital system comply with the requirements of IEEE 289-1971 and satisfy the requirements of GDC 22.

The staff concludes that the ESFAS is designed with proper consideration of safe failure states or other acceptable states for conditions such as disconnection of the system, loss of energy, or where a postulated adverse environment is experienced. Consequently, the staff concludes that the ESFAS satisfies the requirements of GDC 23.

The staff reviewed the design of the ESFAS and plant operating control systems, found that no portion of the EFAS is used for both protective and control functions and signals sent from EFAS to other systems are isolated through fiber-optic cables for signal transmission. Therefore, the staff concludes that the ESFAS satisfies the requirements of IEEE 279-1971 with regard to control and protection system interactions. The staff concludes that the ESFAS satisfies the requirements of GDC 24.

The staff concludes that the ESFAS design satisfies the requirements of 10 CFR 50.55(h) and IEEE 279-1971.

The staff evaluated the ESF-CCS for conformance to the requirements for testability, operability with both onsite power and offsite power, and single failures consistent with the GDC applicable to these systems. The staff concludes that the ESF component control systems are independent and satisfy the single-failure criterion and, therefore, meet the relevant requirements of GDC 34, 35, 38, and 41.

The staff reviewed the dependence of the ESFAS and ESF-CCS on the availability of the essential auxiliary supporting systems and concludes that the designs of the ESFAS and ESF-CCS are compatible with the functional performance requirements of the supporting systems. The staff, therefore, concludes that the interfaces between the design of the ESFAS and ESF-CCS and the design of the essential auxiliary supporting systems are acceptable.

All portions of the ESFAS and ESF-CCS designs are acceptable. The detailed design of the digital I&C system is controlled by the Software Program Manual and verified by the relevant ITAAC (Section 7.1). ABB-CE resolved

the possibility of CMF in digital systems and acceptable diversity in digital systems. The staff reviewed ABB-CE's diversity analysis, and found the analysis to be acceptable.

ABB-CE addressed multiplexer vulnerabilities by (1) evaluating high-quality multiplexers that meet the requirements of industrial standards, (2) assessing vulnerabilities to CMFs, and (3) ensuring adequate defense-in-depth by considering the use of hardwired essential systems controls that bypass the multiplexer networks. The staff reviewed ABB-CE's commercial grade item dedication processes and found the processes to be acceptable.

ABB-CE included sufficient details on the location, inputs (loss of voltage, degraded bus voltage, and A/D conversion), and number of DLSs for the DLS. Consequently, the staff found the DLS hardware and software acceptable.

The staff evaluated the quality and diversity of the EFAS and concludes that ABB-CE specified the appropriate quality requirements, and included adequate defense-in-depth and diversity for postulated CMFs. The staff determined that an unreviewed safety question would result from any changes to commitments for (1) the ESFAS hardware and SDP and its implementation and quality standards, (2) the commercial-grade dedication process for processors and multiplexers, and (3) the program and standards for equipment qualification. Therefore, any change to these issues requires NRC review and acceptance before being implemented. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after the COL is issued.

7.4 Systems Required for Safe Shutdown

7.4.1 Discussion

The I&C systems required to place and maintain the reactor in a safe-shutdown condition also frequently support normal plant operations and, therefore, cannot be exclusively associated with the safe-shutdown function. Safe-shutdown conditions are those conditions in which the reactor cannot achieve criticality in violation of the TS, and under which there is an adequate heat sink so that design and safety limits are not exceeded.

The I&C for these systems are included in the ESF-CCS, and the process-CCS.

Only safety-grade systems may be credited for achieving safe-shutdown conditions. These systems must be suitably diverse and redundant so that the systems can perform

their functions assuming a single failure. These systems must be operable from the MCR with either onsite or offsite power. Finally, the systems must be capable of bringing the reactor to cold-shutdown conditions within a reasonable period of time after a limiting single failure.

The acceptance criteria for the review of the systems required for safe shutdown follow:

- GDC 2, "Design Bases for Protection Against Natural Phenomena"
- GDC 4, "Environmental and Missile Design Bases"
- GDC 13, "I&C"
- GDC 19, "Control Room"
- GDC 34, "Residual Heat Removal"
- GDC 35, "Emergency Core Cooling"
- GDC 38, "Containment Heat Removal"

The RGs that support the above requirements follow:

- RG 1.53, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems" (IEEE 379-1972)
- RG 1.62, "Manual Initiation of Protection Actions"
- RG 1.75, "Physical Independence of Electric Systems" (IEEE 384-1974)
- RG 1.105, "Instrument Spans and Setpoints" (ISA 67.04-1982)
- RG 1.118, "Periodic Testing of Electric Power and Protection Systems" (IEEE 338-1977)

7.4.2 Emergency Feedwater System

The EFWS does not operate uniquely during shutdown operations. The staff reviewed the EFWS as described in Section 7.3.

7.4.3 Atmospheric Dump System

The atmospheric dump system removes decay heat from the primary coolant system through the steam generators when the main condenser is not available or the MSIVs are closed. The decay heat is removed by venting steam to the atmosphere through safety-related modulating ADVs on each of the four main steam lines outside the containment building and upstream of the MSIVs. The ADVs are of a size to allow a controlled plant cooldown in the event of a main steamline isolation, concurrent with a loss of normal ac power and a single active failure of the remaining ADVs.

ABB-CE states that the ADVs are electric motor-operated valves that are manually operated from the MCR or the RSP. The valves are designed to close on a loss of ac electrical power or a loss of the control signal, however, the motive power that closes the valves was not described and was designated DSER Open Item 7.4.3-1. In Amendment N to CESSAR-DC, ABB-CE stated that the valves are designed with a return spring, which causes the valve to close on a loss of motive power or a loss of the control signal. The use of this passive device to ensure ADV closure is acceptable since it causes the valve to fail into a safe state. On this basis, DSER Open Item 7.4.3-1 is resolved.

The ADVs may be positioned using a jog circuit. If offsite power is lost, the ADVs are powered from the DGs. To meet single-failure criteria for a main steamline break, the set of ADVs on each generator are powered from different DGs. If the DGs are not available, the ADVs may also be powered from the gas turbine generator (the alternate power source), or from the Class 1E batteries through inverters, with each ADV powered from a separate channel.

The normally open isolation valve upstream of each ADV is locked open from the MCR and can be remotely and manually positioned from the MCR and from the RSP. The design includes valve position indication (both analog position and open/closed indication lights) in the MCR and at the RSP.

DSER Open Item 7.4.3-2 required clarification of the location of handwheels for operating the ADVs. ABB-CE stated that each ADV may be remotely operated from the MCR or RSP, and includes accessible handwheels for local operation after a loss of the normal power supply. This resolves DSER Open Item 7.4.3-2 regarding the location of the handwheels. The ADVs and operators are designed to meet the requirements of IEEE 279-1971, IEEE 603-1980, and IEEE 308-1980 for the need to design against single failures. The staff finds the ADV control design to be acceptable.

7.4.4 Shutdown Cooling System

The SCS reduces the RCS temperature after shutdown from the hot-shutdown operating temperature to the refueling temperature, and maintains this temperature during refueling. The SCS also cools the RCS coolant after a small-break LOCA, steam and feedwater line breaks, and steam generator tube ruptures, and during startup to maintain flow through the core before the RCPs are started. The SCS also protects against low-temperature overpressure conditions.

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The SCS consists of two independent divisions, with Class 1E power sources for their actuated equipment. The SCS isolation valve interlocks are actuated through the ESF-CCS so that a single failure will not prevent actuation or cause inadvertent actuation of the SCS. The safety-related components of this system can receive power from normal offsite power sources and from physically and electrically independent and redundant emergency power sources.

The SCS is controlled with dedicated redundant instruments that enable operators to monitor the RCS cooldown rate and shutdown cooling flow rate from the MCR, RSP, and local control stations.

The failure position of each valve on loss of actuating signal or power supply is selected to ensure safe operation. Valve position is indicated in the MCR. A momentary pushbutton, with appropriate status control on the MCP, a manual override handwheel, or both, ensure safe plant operation.

In the earlier versions of the CESSAR-DC, ABB-CE stated that an interlock prevents the suction isolation valves from opening when the RCS pressure exceeds the SCS design pressure. Alarms annunciate when the SCS suction isolation valves are not fully open (concurrent with low RCS temperature) to protect against low-temperature overpressure events. Additionally, if the isolation valves are open and the RCS pressure exceeds the setpoint, an alarm in the MCR will annunciate this condition. In the DSER, the staff stated that if this alarm is required for the operator to isolate the SCS to prevent over-pressurizing the SCS, then the alarm must be Class 1E. The low-temperature overpressure protection (LTOP) arming alarm must also be Class 1E if automatic arming does not occur. These issues were designated DSER Open Item 7.4.4-1. ABB-CE later stated that each SCS train includes relief valves downstream of the isolation valve. Either valve has sufficient setpoint and flow capacity to prevent the RCS pressure from exceeding the LTOP criteria for the worst-case overpressure transient during low-temperature operation. Each separate division of the ESF-CCS (A, B, C, and D) provides control, indication, and alarm initiation for each of the four valves. Therefore, since each of the two relief valves provides full capacity for this protective function, both the system and the I&C equipment have full redundancy for this success path. The SCS will withstand such pressure transients. ABB-CE concludes that a Class 1E alarm is not required because the overpressure protection automatically provided by the relief valves eliminates the need for the operator to act for this protective function.

ABB-CE stated that plant procedures will specify the opening of the SCS relief (isolation) valves, to ensure RCS

overpressure protection for all temperatures below the LTOP temperature. These temperature displays are Class 1E. These displays are part of the minimum set of displays relied on by the operator, as stated in Chapter 18. Section 3.4.11 of the TS includes surveillance requirements to verify that the isolation valves are open.

Because of the SCS relief valves and the Class 1E temperature displays as discussed above, the staff concludes that the SCS LTOP alarms are not required to be Class 1E. The COL applicant will verify all relevant controls and displays in the ITAAC. DSER Open Item 7.4.4-1 is resolved.

In a request for additional information (RAI) Q500.11, the staff asked ABB-CE to discuss measures proposed for protection against bypass of the SCS suction line valve interlocks and manipulation of the valves by a single knowledgeable insider at the valve motor control centers. In its letter dated September 28, 1989, ABB-CE explained the design measures that will inherently protect against setpoint manipulation and explained that the security access control system will protect the motor control centers.

Although, as noted in CESSAR-DC Appendix 13A, this is not as great a concern for System 80+ because the SCS system piping has a nominal rating of 6200 kPa (900 psia), ABB-CE committed to amend CESSAR-DC Section 7.6.1.1.1 and Appendix 13A, Sections 7 and 8 to reflect that response. This was designated DSER Confirmatory Item 7.4.4-1. ABB-CE amended the CESSAR-DC to reflect the design and to discuss the features of the system that address sabotage. The features of the design that address sabotage are discussed in Chapter 13. The features of the design that address the I&C aspects, such as the fail-safe features and the design against single failure, are acceptable since they meet the requirements of IEEE Std 279 and GDC 23.

7.4.5 Safety Injection System

The SIS provides highly borated cooling water to the RCS to achieve cold-shutdown conditions upon receipt of a SIAS. During the hot-shutdown and cold-shutdown phases of operation, the SIS may be used to add boron to the reactor coolant if the chemical and volume control system (CVCS) is not available.

A SIAS initiates operation of the SIS in the event of low RCS pressure (derived from four independent pressurizer pressure sensors) or high containment pressure (derived from four independent containment pressure sensors). These parameters indicate the occurrence of a loss-of-coolant accident. A trip from any two of the four sensors will initiate an SIAS. An SIAS also may be manually

initiated from the MCR or from the ESF-CCS equipment cabinets. The SIAS is described in Section 7.3.

The SIS components actuate in the following sequence to aid in achieving cold shutdown:

1. the SIS pumps and SIS pump discharge valves are operated in coordination to adjust and maintain the correct pressurizer water level
2. the boron concentration is periodically sampled and adjusted to compensate for the temperature decrease and other variables until the shutdown concentration is achieved

The pressurizer level is automatically controlled during normal operation by the pressurizer level control system discussed in Section 7.7. Section 7.3 discusses the operation of the SIS for RCS inventory control. Boric acid is injected to ensure that there is a sufficient shutdown margin as the RCS is cooled down. Section 7.7 discusses the boron dilution alarm logic (BDAL), which detects inadvertent boron dilution events while the plant is in Modes 3 through 6. Control board process indicators and status instruments enable the operator to evaluate system performance and manually control system operation. Section 7.3 discusses the interlocks, sequencing, bypassing, redundancy, diversity, and power supplies.

In addition to the controls and indications in the MCR, there are controls outside the control room at the RSP and in the equipment rooms for bringing the reactor to hot standby, maintaining the unit in a safe condition during the hot shutdown, and achieving cold shutdown. The RSP enables the operator to control SIS pumps 3 and 4, the SIS header 3 and 4 valves, pressurizer reactor coolant gas vent (RCGV) valves, the EFWS and ADVs, and the pressurizer sprays and heaters, and gives indications of the systems status. The SIS satisfies the criteria in IEEE 279-1971 as discussed in Section 7.3.9.4. The display of the pressurizer RCGV position satisfies the requirements of 10 CFR 50.34(f)(2)(xi) of providing direct indication of relief valve position in the control room.

7.4.6 Emergency Shutdown From Outside the Control Room

The System 80+ design includes a RSP outside the MCR to facilitate plant shutdown if the MCR must be abandoned. The RSP provides the operators with the capability to bring the plant to hot-standby conditions, maintain the plant in hot shutdown, and achieve cold-shutdown conditions. The instruments in the RSP are listed in the corresponding systems sections of this report.

The RSP gives the operator a minimum of two isolated, redundant channels of safety-related instruments and controls necessary to achieve and maintain hot standby conditions in the reactor (subcritical at operating pressure and temperature), and to achieve cold shutdown of the reactor through the use of suitable procedures. The staff reviewed the RSP instrumentation and controls and concludes that the RSP design is acceptable since the instrumentation and controls at the RSP meet GDC 19.

The MCR and RSP are physically separate, with separate ventilation systems, multiple communications systems, and lighted access between the RSP and the MCR. The design includes signal isolation and disabling of all main controls and the transfer of all hot-standby controls to the RSP. The RSP control systems are fiber optically isolated from the MCR. The staff concludes that this isolation is acceptable since it enables the equipment at the RSP to operate without interaction, independent of the equipment in the main control room.

The RSP controls are activated by MTSs on the ESF equipment cabinets in the ESF equipment room, and at the exits from the MCR. This was designated DSER Confirmatory Item 7.8.8-1.

There was insufficient design information in CESSAR-DC for the staff to determine the adequacy of the MTS design. The lack of detailed MTS information was designated DSER Open Item 7.3.1-2. ABB-CE submitted additional information to close this open item in Amendment N.

One transfer switch for each division of ESF-CCS (four divisions) and each division of the process-CCS (two divisions) is at each exit from the MCR and in the corresponding division equipment rooms. Actuation of the six switches transfers control of the four ESF-CCS divisions and the two process-CCS divisions from the MCR to the RSP. The control is performed through a software-based microprocessor, which deactivates the MCP as a control interface and activates the RSP control interface. The use of software for effecting the control transfer is acceptable. The staff reviewed the computer development process, including software issues, and the acceptance of computer-based system is described in Section 7.1. The staff will audit ABB-CE's software development process and associated ITAAC to confirm the proper implementation of the MTS. Any changes to these commitments would involve an unreviewed safety question and, therefore, will require NRC review and acceptance before being implemented. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

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Transfer initiated by these switches cannot be reversed to the MCPs except through controls at the maintenance and test panels for each division of the ESF-CCS and process-CCS in the equipment rooms for each channel. The staff finds acceptable this method of isolation and independence since it enables the equipment at the RSP to operate without interaction, independent of the equipment in the main control room.

The DIAS and DPS at both the MCP and the RSP display the status of a control transfer. Each system provides an alarm for each division in which the transfer logic has transferred control to the RSP. The component controls within each division also report component group transfer status to the DIAS and DPS. Illuminated pushbuttons and control displays at the manual control interface that have lost control capability due to the transfer will indicate the transfer by not responding. The staff finds acceptable the method of alerting the operators that the transfer has been performed since it meets the IEEE Std 279 requirement of providing the operator with information pertinent to the status of systems required for safe shutdown.

The fiber-optic transfer switches and cables maintain isolation between the ESF-CCS divisions and between the process-CCS divisions. No direct electrical connection exists between the switches and the ESF-CCS, the process-CCS, or the MCP. The fiber-optic switches and cables are an acceptable method of isolation since they meet the IEEE Std 279 requirement of preventing credible failures at their output from affecting the minimum performance requirements of the associated safety systems.

Figure 7.4 shows the transfer switches for one division of the ESF-CCS. The transfer logic for each division receives signals from three locations: MCR exit 1, MCR exit 2, and the maintenance and test panel for the division. The logic transfers the human-machine control interface for the ESF-CCS components controlled by the ESF-CCS for the division. A bridge between the data networks in the ESF-CCS and PPS channel for the division transmits the transfer signal to effect the transfer of the interface for manually sending the PPS channel signals for ESF actuation. The interface for manual initiation of reactor trip is not transferred; manual reactor trip can be performed from either the MCP or the RCP at any time. The staff finds this design acceptable since it meets GDC 19.

Figure 7.5 shows the transfer switches for one division of the process-CCS. The process-CCS MTS functions in the same manner as the ESF-CCS MTS, and is acceptable since it meets GDC 19.

Since the automatic actuation signals are not deactivated by the MTS, testing may be performed by actuating the MTS

switch for a division without deactivating the automatic actuation features of the division. That is, only the capability for manual control is transferred. The staff finds the test provisions of the MTS design acceptable.

The staff concludes that the I&C system design at the RSP and the interfaces at the MCR satisfies the criteria for channel integrity, channel independence and single failure of IEEE 279-1971. On this basis, Confirmatory Item 7.8.8-1 and DSER Open Item 7.3.1-2 are resolved. Chapter 3 addresses conformance to GDC 19.

7.4.7 Safety Depressurization System

The SDS allows the operator to vent non-condensable gases from the reactor vessel upper head and the pressurizer upper head, rapidly reduce the RCS pressure in response to selected severe accident scenarios, slowly depressurize the RCS when the pressurizer main spray and auxiliary spray systems are not available, and remove decay heat through feed-and-bleed operations.

The SDS consists of vent lines from the reactor vessel upper head and the pressurizer upper head. These lines contain redundant RCGV valves that discharge to the reactor drain tank, and redundant pressurizer bleed valves that discharge to the IRWST.

The RCGV valves and bleed valves are designed to be operable following a loss of normal ac power. The valve operators receive power from two independent power trains and can be powered from the EDGs and the alternate power source. The RCGV valves may be opened from the MCR and the RSP.

The valve trains are designed so that a single active failure cannot prevent the establishment of a vent path between the reactor vessel upper head and the pressurizer upper head and cannot prevent isolation of the pressurizer upper head from the reactor vessel upper head. The RCGV valves fail closed. The bleed valves fail in the existing position.

The rapid depressurization (bleed valve) flow paths are manually controlled from the MCR and remote shutdown room (RSR) to allow primary pressure to be reduced below the initiation pressure of the SCS. Operators can remotely control the flow by modulating the bleed valves, but only during a beyond-design-basis event. The valve positions displayed in the MCR and at the RSP satisfy the requirements of 10 CFR 50.34(f)(2)(xi) for providing direct indication of relief valve position in the control room.

Two Class 1E ac buses supply the electrical power required for the RCGV valve and bleed valve control.

Figure 7.4. Interface diagram for Division A master transfer switches.

Figure 7.5. Interface diagram for Process-CCS Division X master transfer switch

The controls can also receive power from the EDGs and the alternate power source.

The staff concludes that the SDS meets the requirements of IEEE 279-1971, and is designed in accordance with RGs 1.75 on physical independence of electric systems and 1.47 on bypass and inoperable status indication.

7.4.8 Station Service Water System

The SSWS transfers heat released from plant systems components and structures to the ultimate heat sink (UHS). The SSWS receives water from the UHS to cool the water in the closed component cooling water system (CCWS), which cools the essential and nonessential reactor auxiliary loads.

Each of the four identical station service water pumps, grouped two in a division, can provide 100 percent of the required flow for conditions after a LOCA. During normal operation, only one pump per division is required to be operating. The second pump in each division automatically starts on a sensed low pump discharge pressure, which indicates the operating pump has failed.

The pump motors receive power from their associated 4160-V Class 1E auxiliary power system. If offsite power is lost, the pumps are stopped and restarted in accordance with the emergency DG load sequencing. The control to start and stop each pump is in a separate channel. When a station LOOP occurs, the emergency load centers can power only one SSW pump in each division. The operator can add a second divisional pump if enough loads have been removed from the emergency load center.

Both divisions and all four station service water pumps are started by either a manual start signal by a control room operator or a low station service water pump discharge pressure signal. A manual start or stop signal from the control room overrides the automatic mode. Manual start and stop controls for each SSWS division permit the operator to remove a division from operation after automatic operation actuation, if that division is not required. The only components actuated manually by the control room operator are the station service water pumps, the strainer backwash system, and the isolation valves for the SSWS component cooling water heat exchanger. Valves (except check valves) in the supply lines from the pumps and in the return lines to the UHS are either locked open or locked closed depending on plant status and requirements.

After an ESFAS, the operating SSWS pumps continue running using normal power. If one division is operating, a LOOP coincident with an ESFAS will cause the operat-

ing SSW pumps to shut down and then be sequenced onto the EDGs. If two pumps in one of the divisions are operating before this event, the DG sequencers will load only one SSW pump for each division. Operators retain manual control of the SSW pumps during emergency conditions.

The SSW instruments allow the system to operate automatically by remote control, and give continuous indication of system parameters (UHS water temperature, station service water pump flow rate, UHS water level) locally, in the MCR, and in the RSR. Process indications and alarms enable the operator to evaluate the SSWS performance and to detect malfunctions. Sensors monitor the discharge pressure of the station service water pump and actuate alarms upon detecting an abnormally low or abnormally high pressure. The UHS water levels and temperatures are monitored to detect a low or high level, or a high-temperature condition. The SSW discharge temperature from the CCWS heat exchangers is monitored (with an alarm to the control room) to detect reduced water flow to the heat exchanger or an abnormal heat supply into the heat exchanger from the CCWS.

Differential flow between the station service water pump discharge (total for each division) and component cooling water heat exchanger discharge (total for each division) is indicated to reveal leakage. A high differential flow alarm indicates a significant line break. A radiation monitor in each division detects leakage of radioactive component cooling water into the SSWS.

The staff reviewed the I&C in the MCR, RSP, and the local control stations. The instrumentation, controls, and alarms are acceptable for determining SSWS status per GDC 19, and satisfy the criteria of IEEE 279-1971.

7.4.9 Component Cooling Water System

The CCWS is an intermediate cooling water system between the RCS and the SSWS. The CCWS transfers the heat from the RCS closed system to the UHS, which is the source of the SSWS cooling water.

The CCWS supplies cooling water to the RCPs, letdown heat exchanger, SCS heat exchanger, core spray system heat exchanger, sample coolers, nonessential chillers, and other nonessential reactor auxiliary cooling loads.

The CCWS instrumentation and controls enable the operator to control the system from the MCR, RSP, and at local stations.

The CCWS provides discharge pressure indications for the sump pumps, CCWS pumps, heat exchangers, and chemi-

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cal addition tank vents and discharge lines. Plant personnel can perform maintenance and tests at test points throughout the CCWS. CCWS controls will automatically start the corresponding redundant CCWS pump in each division on low CCWS pump differential pressure and will open the containment spray heat exchanger isolation valve on CCWS pump high differential pressure.

The CCWS sends temperature indications to the MCR and RSP for the CCWS heat exchanger inlets and outlets. The CCWS controls allow the operator to bypass the CCWS heat exchangers to maintain the required minimum temperature at the heat exchanger outlet. The operator may also regulate CCWS temperature by controlling the letdown heat exchanger outlet temperature and charging pump mini-flow heat exchanger control valves.

The CCWS sends indications of outlet flows from the CCWS heat exchangers and CCWS pump discharge flows to the MCR and the RSR. Flow indications are provided at local test points to facilitate system maintenance.

The MCR and RSP include level indications for CCWS surge tank, sump, heat exchanger structure sump, and component cooling chemical addition tank. The CCWS sump pumps automatically start on high sump level, and automatically stop on low level.

Radiation monitors downstream of the CCWS pumps alert the operators when radioactivity levels exceed a preset level above background. The CCWS radiation levels are indicated in the MCR and the RSR.

The staff reviewed the I&C in the MCR, the RSP, and the local control stations. The instrumentation, controls, and alarms are acceptable for determining CCWS status per GDC 19, and satisfy the criteria of IEEE 279-1971.

7.4.10 Onsite Power Systems

The staff reviewed the onsite power system as presented in Chapter 8.

Two independent, 100-percent-capacity DGs provide emergency power to start and supply the essential loads to shut the plant down safely and to maintain the plant in a safe condition following a loss of offsite power. Load sequencing is a part of the ESF-CCS, and is described in Section 7.3.

The DGs are started automatically upon receiving an ESFAS or SIAS indicating an undervoltage (resulting from a LOOP on the associated 416-V ESF bus).

The unit design includes two redundant 4,160-V Class 1E buses, four redundant 480-V Class 1E buses, and four redundant 120-V Class 1E auxiliary power buses. One system can be taken out of service for testing during reactor operation. The breaker racks can be moved to the test position while the system is tested. The MCR displays continuous indication of the unavailability of the racked-out system. The Class 1E auxiliary power system allows for inspections, testing, and maintenance without affecting the operation of the unit. This complies with GDC 18 and RG 1.22.

The protective relays will be tested periodically in a testing facility that meets the GDC 18 criterion and RG 1.22. Relay sensors such as current transformers will be tested before initial installation and unit operation, and periodically thereafter. These protective devices are in service during normal operation. The preoperational tests for the protective relaying system must verify the continuity of the system and the condition of all the components. The methods to accomplish these tests follow:

- Test all relays and other momentary duty-type operating devices for the protective relaying of the onsite power system to determine individual performance characteristics, and to verify that design settings can be repeated under various simulated conditions. The intent of these tests is to show design integrity.
- Test all relay sensors, such as current and potential transformers, for correct and reliable output.
- Inspect all interconnecting wires and cables for proper installation and connections.
- Test all protective relaying systems under necessary simulated conditions to verify correct operation in preferred, alternate, and abnormal modes.

Inspection, maintenance, and testing of the Class 1E dc systems will be performed periodically in accordance with RG 1.22. The periodic testing program will be scheduled so as to not interfere with plant operations.

A bypassed or deliberately induced inoperable status will be indicated automatically for Class 1E power systems required for safety. This commitment satisfies RG 1.47 on bypass and operable status indication.

The routing of 1E and associated electrical cabling and sensing lines from the sensors is designed to meet the guidelines of RGs 1.51 and 1.75. The cabling is routed to minimize the possibility of CMFs. The cabling for the four safety channels is routed separately; however, the cables of various functions within the same division may

be routed together. Low-energy signal cables are routed separately from other power cables. Safety-related redundant sensors are separated. The separation of safety-related cables requires that the cables be routed in separate cable trays. Associated circuit cabling from redundant channels is handled the same as Class 1E cabling.

Cabling for redundant channels of safety-related circuits is installed so that a single failure will not cause multiple channel malfunctions or interactions between channels.

Other instrumentation circuits and cables that may be near Class 1E or associated circuits and cables are treated as associated circuits unless analyses or tests demonstrate that credible failures of the cabling cannot adversely affect Class 1E circuits. Non-Class 1E channels X and Y I&C circuits and cables are routed separately from each other.

The TS specify periodic testing requirements of the electric power and protection systems. Chapter 16 discusses the TS. This resolved DSER Open Item 7.4.10-1 (TS had not been submitted for review when the DSER was issued and, therefore, the entire TS review was an open item. The I&C portion of it was designated Open Item 7.4.10-1).

The staff review of onsite power system is discussed in Section 8.3.

7.4.11 Conclusions

The staff reviewed the systems required for safe shutdown, including the instruments, initiating circuitry, logic elements, interlocks, redundancy features, and actuated devices that perform the I&C functions that remove residual heat from the core, containment, and other vital components and systems, and prevent the reactor from returning to criticality.

The systems required for safe shutdown are acceptable and meet the relevant requirements of GDC 2, 4, 13, 19, 34, 35, and 38. This conclusion is based upon the following:

1. The staff confirmed that these systems conform to the applicable RGs and industry codes and standards.
2. The staff confirmed ABB-CE's choice of systems and components required for safe shutdown that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The systems and components chosen satisfy this aspect of GDC 2 and 4.
3. The staff reviewed the shutdown I&C systems and concludes that they are sufficient to maintain the variables and systems that can affect the fission process, integrity of the core, reactor coolant pressure boundary, and containment and its associated systems within the prescribed operating ranges during plant shutdown operations. The systems required for safe shutdown satisfy the requirements of GDC 13.
4. The instruments and controls in the control room will allow actions to be taken to maintain the plant in a safe condition during shutdown, including a shutdown after an accident. The RSP equipment is appropriately located outside the control room and can promptly bring the reactor to hot shutdown. This equipment includes I&C to maintain the unit in a safe condition during hot shutdown of the reactor, and enables the operators to bring the reactor to cold shutdown by using suitable procedures. Therefore, the systems required for safe shutdown outside the MCR satisfy the requirements of GDC 19.
5. In its review of the systems required for safe shutdown the staff included the dependence of these systems on the availability of essential auxiliary supporting systems. The essential auxiliary supporting systems are compatible with the functional performance requirements of the I&C systems. Therefore, the safe-shutdown systems and the essential auxiliary supporting systems will work together in an acceptable manner.
6. In reviewing the I&C systems required for safe shutdown, the staff evaluated the conformance of the requirements for testability, operability with onsite and offsite electrical power, and single failures consistent with the GDC for safe-shutdown systems. These systems can be tested and can operate with either onsite or offsite electrical power sources. The controls for redundant safe-shutdown systems will be independent and will satisfy the requirements of the single-failure criterion and, therefore, will meet the relevant requirements of GDC 34, 35, and 38.

7.5 Information Systems Important to Safety

7.5.1 Discussion

The staff reviewed the safety-related instruments available to the operator for monitoring plant systems conditions and for performing manual actions important to plant safety.

The System 80+ control room is designed to provide safety-related and non-safety-related information and control capabilities that the operator can use for both normal operations and accident mitigation. The design objective is to minimize the operator's transition from normal plant operations into accident recovery operations. The information systems also include data validation to

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ensure the operator is presented with unambiguous indications of the plant state.

Plant state information is presented to the operator in the following information display systems:

(1) Integrated Process Status Overview Panel

The integrated process status overview panel (IPSO) is a large panel that depicts a cursory overview of the state of major plant systems and safety functions. The IPSO is located at the front of the control room.

(2) Discrete Indication and Alarm System

Discrete indicators display validated safety and non-safety plant process parameters. The DIAS also generates and presents the alarm states in an integrated manner through alarm indications, message displays, and DPS video displays.

(3) Data Processing System

The DPS acquires and validates plant data and presents the resulting information using a hierarchical format on video display pages.

(4) Component Control and System Operator Module Displays

The MCP incorporates displays of on or off, open or closed status from other I&C system operator modules and plant components.

Section 7.7 describes the IPSO, DIAS, and DPS in the System 80+ design.

7.5.2 System Description

7.5.2.1 Safety-Related Plant Process Display Instruments

The process instruments in SAR Tables 7.5-1 through 7.5-3 give plant status information to the operator in the control room. The staff reviewed this information against RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident." The information presented in the three tables cited above conforms to the guidelines presented in RG 1.97; however, ABB-CE did not discuss Type A variables in the SAR prior to issuance

of the DSER. This was designated DSER Open Item 7.5.2.1-1.

RG 1.97 defines Type A variables as those variables that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design-basis events. ABB-CE later stated it did not list Class 1E alarms or Type A variables because the System 80+ performs all protective functions using automatic or passive responses, without the need for operator action to accomplish the protective function. SAR Chapter 18 lists the minimum set of displays and alarms required by the emergency procedure guidelines. The staff finds acceptable ABB-CE's selection of indications for monitoring postaccident conditions. The list of these variables satisfies the requirements of 10 CFR 50.34(f)(2)(xvii) (TMI Action Item II.F.1) and 10 CFR 50.34(f)(2)(xix) (TMI Action Item II.F.3). This resolves Open Item 7.5.2.1-1.

The staff reviewed the safety-related process display instruments and the ESF system parameters and found sufficient redundancy and adequate ranges for these instruments. The ranges given in the SAR are typical ranges. ABB-CE stated that the actual ranges will be based on the equipment to be procured at each site. Therefore, ABB-CE indicated that the COL applicant should make appropriate adjustments for the actual components. ABB-CE revised the CESSAR-DC tables to state that the Indicated Range column in CESSAR-DC Table 7.5-1 would be relabeled to state that the values in this column are the Minimum Indicated Range. The staff reviewed these ranges against Table 3 of RG 1.97 and found them acceptable. This resolves DSER Open Item 7.5.2.1-2. These ranges satisfy the requirements of 10 CFR 50.34(f)(2)(xix) (TMI Action Item II.F.3).

Alternate information for startup, operation, and shutdown is displayed at the RSP and local stations outside the control room to allow the reactor to be shut down and maintained in a safe condition if the control room becomes uninhabitable. The staff accepted this information as discussed in Section 7.4.

7.5.2.2 Reactor Trip System Monitoring

Even though the RTS is automatic and does not require operation action (except for a manual trip capability), the operator in the control room receives information to confirm that a reactor trip has taken place and to confirm whether a limiting safety system setting has been reached. This information consists of indications of

- process parameters that indicate reactor trip
- trip, pre-trip, and bypass lights
- audible and visual alarms
- CEA "dropped rod" information
- trip switchgear circuit-breaker position

Operating bypass is indicated on the remote modules located in the MCR, as described in Section 7.1. Individual trip channel bypass is locally indicated at the PPS and on the remote modules in the MCR. The DIAS and DPS display CEA rod position, dropped rod, audible and visual alarms, and trip switchgear circuit-breaker position. These indications satisfy the requirements of 10 CFR 50.34(f)(2)(xix) to provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

7.5.2.3 Engineered Safety Features Monitoring

The ESFAS continuously monitors the system input parameters, using actuation logic to initiate the ESF systems when these parameters reach their trip setpoints. After actuation, the ESF systems continue to function for up to 30 minutes without requiring operator action. Operator action may then be required to start other systems such as the SCS.

The operator in the control room receives information from the DPS to monitor the ESF and related systems after an event. This information consists of valve position indication, pump operating status, tank level indications, flow indications, indications of the process parameters that actuate ESF systems, and ESF system performance displays. Four remote modules indicate the pretrip, trip, bypass, and operating bypass condition for each of the associated actuation system input signals. The bypass of an individual trip channel is indicated at the PPS cabinet and the modules in the MCR. The valve position indications satisfy the requirements of 10 CFR 50.34(f)(2)(xi) to provide direct indication of relief and safety valve position in the control room, and 10 CFR 50.34(f)(2)(xix) to provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

7.5.2.4 Control Element Assembly Position Indication

The pulse-counting CEA position indication system and the reed-switch CEA position-indication system are two diverse, independent CEA position-indication systems that give CEA position information to the operator.

The CEDMCS receives automatic CEA motion demand signals from the reactor-regulating system (RRS) or manual motion signals from the CEDMCS operator's

module and converts these signals to dc pulses that are transmitted to the control element drive mechanism (CEDM) coils to cause CEA motion. The CEDMCS counts these pulses to infer the CEA position by electronically monitoring the mechanical actions within each CEDM to determine when a CEDM has raised or lowered the CEA. Section 7.7 discusses the pulse-counting system.

A series of magnetically actuated reed-switch position transmitters (RSPTs) in the reed-switch CEA position-indication system send signals representing CEA position. Two independent reed-switch assemblies monitor each CEA. The RSPTs send an analog position-indication signal and three physically separate discrete reed-switch position signals. The analog position-indication system includes a series of magnetically actuated reed switches spaced at 3.8-cm (1.5-in.) intervals along the RSPT assembly with precision resistors between the switches. These switches form a voltage divider network. The RSPT is adjacent to the CEA extension shaft and actuating magnet. The analog output signal is proportional to the CEA position within the reactor core.

The three separate discrete position signals are contact closure signals from the three separately located reed switches. These signals are an upper electrical limit (UEL), a lower electrical limit (LEL), and a drop rod contact (DRC). The three discrete CEA position reed switches (UEL, LEL, DRC) give contact closure signals to the CEDMCS for use with the CEA positions in controlling the CEA and displaying position on the main control board.

The RSPTs send analog reed-switch position signals to the CPC (as discussed in Section 7.2). CEA position information passes to the CPCs directly and the CEACs. The CEACs display the position of each regulating rod, part-strength rod, and shutdown rod upon operator demand in a bar chart format on Channels B and C CPC operators modules at the main control board. The operator can also address any analog position signal for display on the CPC operator's module.

The CEACs send CEA deviation information to the CPCs and a CEA deviation alarm. The deviation alarm is generated whenever the difference between the highest and lowest CEA positions in a subgroup exceeds a predetermined allowable deviation. The CPC uses the CEA deviation information in determining power distribution for the low DNBR and high LPD trip functions. Pre-trip alarms are initiated if the DNBR or LPD trip limits are approached. A pre-trip alarm indication appears on the PPS operator's control panel (both local and remote) and is displayed on the DIAS.

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The DPS includes a validated CEA position CRT display, based on the RSPT and CEDMCS pulse-counting CEA position information. The validation logic identifies and alarms discrepancies between these two position-indication signals. These indications satisfy the requirements of 10 CFR 50.34(f)(2)(xix) to provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

7.5.2.5 Postaccident Monitoring

The PAMI allows the operator to assess the state of the plant during and after an accident. Most of these indications monitor instruments, equipment, or systems that respond automatically for the design basis event.

The MCR integrates the safety parameter display system (SPDS) with PAMI as follows:

1. Seismically qualified DIAS-P processors and displays are dedicated to continuously monitor and display RG 1.97 Category 1 parameters. These displays are located in the MCR safety-monitoring panel.
2. Seismically qualified DIAS-N displays are integrated into the MCR panels for normal operations and display RG 1.97 Category 1, 2, and 3 parameters. These displays enable the operator to select various PAMI channels, and are isolated from the DIAS-P displays.
3. The DPS CRT displays all RG 1.97 variables in a manner that is isolated from the DIAS-P and DIAS-N. The DPS provides integrated displays for monitoring critical safety functions, inadequate core cooling (ICC), and other safety-related plant parameters. The DPS also stores, retrieves, and trends historical data. The DPS design includes isolated data links to the onsite TSC and emergency operations facility (EOF) to enable the operator to monitor plant conditions.

The list of process parameters and the corresponding ranges conform to Table 3 of RG 1.97.

7.5.2.6 Automatic Bypass Indication on a System Level

Section 7.1 describes automatic bypass indication on a system level.

7.5.2.7 Inadequate Core Cooling Monitoring Instrumentation

The ICC monitoring system detects the beginning of and recovery from a loss of core-cooling capability. The sensor inputs for the ICC parameters are signal-conditioned by the two-channel PAMI processors and transmitted to the

DIAS and DPS for primary display and trending. The instrument sensors that monitor ICC progression are:

- resistance temperature detectors (RTDs) in the coolant loops
- pressurizer pressure sensors
- reactor vessel level monitors using the heated junction thermocouples
- core exit thermocouples (CETs)

The signals from the RTDs in the hot legs and cold legs, the maximum temperature of the top three unheated thermocouples in the HJTCs, and the minimum pressure from the pressure sensors in the pressurizer are combined in the saturation margin monitors to indicate a loss of subcooling, the occurrence of saturation conditions, and the achievement of subcooling following core recovery.

The reactor vessel level monitors inform the operator of the reactor vessel liquid inventory above the fuel alignment plate, and the progression of core recovery after a core-uncovering event. The HJTC probe assemblies consist of discrete HJTC sensors at different levels ranging from the top of the fuel alignment plate to the reactor vessel upper head. The ICC monitoring system includes two probe assemblies, each consisting of eight HJTC sensors, a separator tube, a seal plug, and electrical connectors. The eight HJTC sensors are electrically independent. The output signals are sent to the DIAS and DPS. DIAS-P generates an alarm signal when any of the HJTCs detects the absence of liquid at any HJTC probe.

The CETs monitor water temperature at the exit of the core. This measurement indicates the amount of core heatup during an ICC event. The CET signals are processed by the PAMI processors to produce a representative core exit temperature. This temperature will be either the maximum valid core exit temperature or a statistically derived value representing 95 percent of the temperature distribution. The DIAS and DPS process the output signal for displays and alarms. DIAS-P generates an alarm when any of the CET temperatures reaches a value that indicates the probability of ICC.

The display of the ICC information is described in Sections 7.5.2.7.1, 7.5.2.7.2, and 7.5.2.7.3.

7.5.2.7.1 DIAS Channel P

DIAS-P processing equipment provides continuous display of the SMM sensors, the HJTC sensors, CETs, and all RG 1.97 Category 1 variables on the safety monitoring panel

in the MCR. These displays are seismically qualified and provide dedicated, validated indication of each ICC-sensed parameter. If the variable cannot be validated by the DIAS algorithm, the condition will be alarmed, and the operator can then select any input channel. Nevertheless, either a validated or individual channel is presented at all times.

DIAS-P also generates an alarm signal when (1) any of the HJTCs detects the absence of liquid or (2) any of the CET temperatures reaches a value that indicates the probability of ICC. The alarm signal is transmitted to the DPS for display.

7.5.2.7.2 DIAS Channel N

The ICC-sensed parameters are also available on DIAS-N displays on the applicable plant system section of the MCPs. These displays are seismically qualified and present validated information.

The DIAS-N processing equipment validates signals from the saturation margin sensors before determining the temperature saturation margin. The SMM is then calculated from the minimum pressure input. The temperature subcooled or superheated margin is based on the difference between the saturation temperature and the sensor temperature input.

Three saturation margins are displayed: RCS saturation margin, upper head saturation margin, and CET saturation margin. The RCS saturation margin is based on the difference between the saturation temperature and the maximum temperature from the RTDs in the hot and cold legs. The upper head saturation margin is based on the difference between the saturation temperature and the unheated junction thermocouple (UHJTC) temperature (based on the maximum of the top three UHJTCs). The CET saturation margin is based on the difference between the saturation temperature and the representative core exit temperature calculated from the CETs.

DIAS-N alarms when the temperature saturation margin reaches a setpoint of 11 °C (20 °F) subcooling [$T_{sat} - 11$ °C (20 °F)] for RCS or upper-head saturation margin. To avoid spurious indications, CET saturation is not alarmed.

7.5.2.7.3 Data Processing System Inadequate Core Cooling Displays

The ICC parameters are incorporated into the displays and alarm logic of the DPS critical function-monitoring (CFM) computer program.

The critical safety functions are directly monitored by a set of algorithms that process the measured plant variables to determine the status of the plant's safety function control. If any of the critical functions exceed their setpoints, the DPS generates a CFM alarm. This critical function alarm logic incorporates signals from the ICC instruments.

The DPS ICC detection displays present an ICC summary page as part of the core heat removal control critical function, supported by more detailed display pages for each of the ICC variable categories.

The summary page includes the maximum of the RCS and upper head saturation margins, the reactor vessel level above the core, and representative core exit temperatures.

The DPS sends the operators additional information through the DPS CRTs in a more comprehensive manner than can be produced by the DIAS-P and DIAS-N processors. All DPS displays are consistent with the DIAS displays.

Saturation margin, HJTC, CET, and ICC are presented in enhanced displays in lower pages of the DPS, which the operator can access from the CRT touch screens.

Section 7.7 describes the validation of this information.

The ICC indications satisfy the requirements of 10 CFR 50.34(f)(2)(xviii) to provide instruments that provide in the control room an unambiguous indication of inadequate core cooling.

7.5.3 Conclusions

The staff reviewed the information systems important to safety to determine their conformance to the acceptance criteria presented in Section 7.5 of the SRP. The following acceptance criteria and guidelines apply to the information systems important to safety:

- GDC 2, "Design Basis for Protection Against Natural Phenomena"
- GDC 4, "Environmental and Missile Design Basis"
- GDC 13, "Instrumentation and Control"
- GDC 19, "Control Room"

GDC 2 and 4 are applicable to variables that are classified as Category 1 and Category 2 in RG 1.97.

The staff used the following guidelines to evaluate the conformance of the system design to the four GDC:

- RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"

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- RG 1.75, "Physical Independence of Electric Systems"
- RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident"
- RG 1.105, "Instrument Setpoints for Safety-Related Systems"
- Position 5 of RG 1.151, "Instrument Sensing Lines"
- RG 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants"
- NUREG-0696, "Functional Criteria for Emergency Response Facilities"

The information systems important to safety display information on the status of the plant to allow the operator to perform manual safety actions when necessary. The staff reviewed tables of system variables and component states to be indicated, and descriptive information about the information systems. The staff also reviewed the acceptance criteria and guidelines listed above. The staff concludes that the information systems important to safety meet the requirements of GDC 2, 4, 13, and 19.

The staff reviewed the choices for systems and components of the information systems that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The displays for Category 1 variables (which include the PAMI parameters) are the DPS, DIAS-N, and DIAS-P. Class 1E qualification includes the entire instrument channel up to the channel isolation device. Class 1E signals are isolated either before transmission to or within qualified I/O sections of the DIAS and DPS. The DIAS displays and processing units are not Class 1E, but are seismically qualified. The DPS also presents Category 1 variables, although it is designed as a non-safety-related system with no functional seismic qualification. Qualification of Category 2 variables is the same as for Category 1 variables, except that display devices not used for Category 1 variables are not seismically qualified.

The DIAS and DPS equipment in the control room and other control building areas are designed and qualified to remain functional after exposures to a temperature range between 23 °C (73 °F) and 30 °C (85 °F), at atmospheric pressure, relative humidity from 23 percent to 100 percent, gamma exposures to 10 Gy (10³ rad), and no exposure to chemical sprays. The temperatures reflect equipment cabinet or panel ambient conditions. An additional 10 °C

(18 °F) is added to normal and abnormal temperatures to account for the increased temperature of subassemblies mounted inside cabinets and panels. The environment in the control building areas is maintained by Class 1E redundant HVAC systems powered from independent Class 1E power sources. The staff initially contended that the DPS and DIAS digital equipment should be qualified for a maximum temperature of 49 °C (120 °F) based on operating experience. This was designated as Open Item 7.5.3-1. ABB-CE stated that the DPS equipment is designed for 40 °C (104 °F), and the DIAS equipment is designed for 55 °C (132 °F). Since the DPS is not part of the safety-related equipment, the lower qualification temperature for the DPS is acceptable. The staff finds acceptable the qualification temperature of the DIAS equipment since it meets the equipment qualification requirement of IEEE Std 279.

ABB-CE did not include sufficient information in the initial CESSAR-DC regarding the monitoring of environmental conditions (e.g., temperature, moisture, humidity, chemical pollutants) and conditions having the potential for functional degradation (e.g., pipe breaks, fires, loss of ventilation, spurious operation of fire protection systems) within the equipment. Operating experience shows that such monitors are necessary to alert plant personnel when equipment is operating in an environment beyond its design limits. This was designated DSER Open Item 7.5.3-2.

ABB-CE later stated that each outside air intake location is monitored for the presence of radioactivity, toxic gases (e.g., chlorine), and products of combustion. The outside air intake is automatically isolated upon indication of radiation, chlorine, or smoke in the air intake. If both air intakes close, the operator can override the intake monitors, and by inspection of the control room indications, select the least contaminated intake.

Instruments for the air conditioning systems indicate the temperature, chlorine, and radioactivity levels. Early warning ionization-type smoke detectors in the supply, return, and outside ductwork serve the control room ventilation system.

Each cabinet containing DIAS and DPS computer equipment has a temperature switch and associated alarm in the MCR to alert the operator if the temperature within the cabinet reaches the upper limit specified for the environment in that location.

Redundant, isolated divisions of safety-related equipment ensure that, in the event of a fire, actuation of the fire suppression system will not affect more than one division of safety-related equipment. Consequently, the effect of fire suppression system water on the electronic components

will be isolated and will not hinder operation of the safety systems.

The staff finds acceptable the means by which ABB-CE has addressed environmental monitoring requirements for the DIAS and DPS since the associated alarms alert plant personnel when the equipment is operating in an environment beyond its design limits. This resolves DSER Open Item 7.5.3-2. The HVAC design is acceptable and discussed in Chapter 9.

ABB-CE committed to qualify equipment in harsh containment environments in accordance with the criteria of IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations"; RG 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants"; and IEEE 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The harsh environment is defined as including temperatures from 43 °C (110 °F) to 204 °C (400 °F); a saturated and superheated mixture of steam and air, radiation TIDs up to 4×10^5 Gy (4×10^7 rad) gamma and 3.5×10^6 Gy (3.5×10^8 rad) beta, and 4,400 ppm boric acid followed by a pH of 7.0-8.5 after 4 hours using trisodium phosphate dodecahydrate. ABB-CE stated that no new harsh environment equipment will be required for the System 80+ design beyond that previously qualified for the System 80 design. This qualification conforms to the requirements of GDC 2 and 4.

The information systems important to safety conform to the guidelines for instruments to access plant conditions during and after an accident, as stated in ANSI/ANS-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors," as supplemented by RG 1.97.

The design includes redundancy for both the instrument channels supplying the signal and for the displays in the control room for Category 1 variables. Instrument channels are electrically independent and physically separated from each other and from non-safety-related equipment by qualified isolation devices. The DIAS-P and DIAS-N displays give credited redundancy for the display of Category 1 variables. These displays are electrically independent and physically separated. The DPS also presents each Category 1 variable to avoid TS limitations for conditions when a DIAS channel is out of service.

The DPS is physically separated and independent of both DIAS channels. Independent Class 1E power busses supply power for each redundant Category 1 sensor instrument channel, up to and including the channel

isolation devices. The DIAS-P processing units and displays are powered from the isolated Class 1E, battery-backed, C and D instrument buses. The DPS is powered from non-safety-related, battery-backed computer buses. The Category 2 variables are displayed on DIAS-N and DPS with power supplies from the non-safety-related instrument buses and computer bus, respectively. Both are battery backed. The instrument channels are powered from the X or Y instrument bus. The redundant information systems conform to the guidelines for the physical independence of electrical systems in RG 1.75.

The ranges of the Category 1 indicators conforms to the ranges specified in RG 1.97. Where the required range of the monitoring instruments results in a loss of sensitivity during normal operating conditions, ABB-CE will install a separate instrument channel. The sensor instrument spans and setpoints conform to the guidelines of RG 1.105.

The staff reviewed the systems for which a bypassed or inoperable status is indicated in the control room. The list of systems includes all of the ESF component control systems and their environmental controls. The staff finds that the bypass indications will give the operators timely information and status so the operators can mitigate the effects of unexpected system unavailabilities. The bypass indications satisfy the guidelines in RG 1.47.

The staff finds that the information systems important to safety conform to the requirements of GDC 13 for monitoring systems and variables over their anticipated ranges for normal operation, for anticipated operational occurrences, and accident conditions. Further, conformance to GDC 13 and the applicable guidelines satisfies the requirements of GDC 19 for information systems in the control room from which actions can be taken to operate the unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

7.6 Interlock Systems Important to Safety

The staff reviewed interlock systems important to safety that operate to reduce the probability of occurrence of specific events or to maintain safety systems in a state to ensure their availability in an accident. These systems include interlock systems to prevent overpressurization of low-pressure systems (e.g., the SCS) when these systems are connected to high-pressure systems (e.g., the RCS), interlocks to prevent overpressure of the RCS during low-temperature operation of the reactor vessel, valve interlocks to ensure the availability of the SITs, interlocks to isolate safety-related systems from non-safety-related systems, and interlocks to preclude inadvertent connections

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between redundant or diverse safety systems where such connections exist for testing or maintenance.

The staff reviewed these systems and confirmed that such design considerations as redundancy, independence, single failures, qualification, bypasses, status indication, and testing are consistent with the design bases of these systems and commensurate with the importance of the safety functions to be performed.

The concerns discussed in Sections 7.1, 7.2, and 7.3 apply to the digital computer systems used in interlock systems.

7.6.1 Shutdown Cooling System Suction Line Valve Interlocks

The SCS is a low-temperature, low-pressure system that removes decay heat from the RCS. The steam generators cool the RCS to approximately 177 °C (350 °F) and 3,103 kPa (450 psia). Below these values, the SCS cools the RCS to refueling temperatures and maintains these conditions for extended periods.

Redundant, motor-driven, interlocked isolation valves on each suction line prevent overpressurization by preventing the suction line isolation valves from being opened if RCS pressure has not decreased below an acceptable value. The pressurizer pressure safety-related instruments are used to display RCS pressure conditions.

The SCS interlocks are redundant so that any single failure will not cause a suction line and heat exchanger to be subjected to pressures greater than design pressure. The interlock cannot be overridden so that operator action cannot inadvertently subject the SCS to RCS pressures. No single failure can prevent the operator from aligning the valves on at least one suction line for shutdown cooling after the RCS pressure requirements are satisfied.

Redundant relief valves on the suction lines prevent or mitigate overpressurization from pressure transients. These transients can be caused by inadvertent starting of SI pumps, charging pumps, inadvertent energization of the pressurizer backup heaters, or a combination of these. The relief valves on each suction line are set to ensure the system pressure remains below the design limit of the SCS. The SCS relief valve positions are displayed in the MCR and at the RSR. These indications satisfy the requirements of 10 CFR 50.34(f)(2)(xi) to provide direct indication of relief valve position.

Interlocks and valves will be tested in accordance with GDC 1 and 21, and RGs 1.22 and 1.68, and the appropriate sections of IEEE 279-1971, 336-1985, 338-1977, and 603-1980.

The method for identifying power and signal cables and cable trays dedicated to the instrument, control, and electrical equipment for the isolation valves will conform to RG 1.75.

The instrumentation, control, and electrical equipment associated with the interlocks will be seismically and environmentally qualified to operate under all required design-basis events in accordance with the seismic and environmental design commitments in SAR Sections 3.10 and 3.11.

7.6.2 Safety Injection Tank Isolation Valve Interlocks

The SIS will inject borated water into the RCS upon receiving an SIAS and will supply long-term cooling in conjunction with other systems after an accident. The SITs inject borated water into the RCS if system pressure decreases below the SIT internal pressure. During normal operation, the motor-operated isolation valve on each tank is open with power removed from its motor circuit to eliminate the possibility of spurious closure. As the RCS pressure is reduced during plant shutdown, the low pressurizer pressure trip setpoint is reduced to avoid inadvertent initiation of SI, the SITs are depressurized to a value below the SCS design pressure, and the valves have their power restored and are closed.

The SIT interlocks prevent the SITs from inadvertently pressurizing the SCS while maintaining the SITs available in case of a LOCA. The isolation valves are manually closed when RCS pressure decreases below 3,391 kPa (475 psig) so that the SITs cannot overpressurize the SCS while the SITs are maintained at some pressure above atmospheric. As RCS pressure increases, the SIT isolation valves will automatically reopen at 4,259 Kpa (600 psig) to ensure that SITs are available for injection during plant startup. If the isolation valves are closed and an SIAS is received, the isolation valves will automatically open. The SIAS overrides the interlock or any manual signal.

7.6.3 DIAS-N and Data Processing System Alarms

The DIAS-N and DPS include alarms for certain operational occurrences for which no specific automatic actuation of a safety system is required. Both of these systems include alarms but DIAS-N activates priority 1 tiles for both systems. Alarms alert the operator to keep the plant operating within TS limits and prevent equipment damage.

The DIAS-N and DPS alarms receive validated signals. The operator cannot bypass these alarm systems and must acknowledge the audible alarms for each channel individually.

7.6.4 Conclusions

The interlock systems important to safety are acceptable and meet the relevant requirements of GDC 2 and 4. The basis for the staff's conclusion is described below.

1. The staff reviewed of the interlock systems important to safety that prevent overpressurization of the SCS when it is connected to the RCS. The staff position for this interlock system is Branch Technical Position (BTP) ICSB-3, "Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System." The staff concludes that the design of this system satisfies the staff's guidelines on interlocks to prevent overpressurization of low pressure systems.
2. The staff reviewed the interlock that prevents overpressurization of the primary coolant system during low-temperature operation. The staff's position for this interlock system is BTP RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures." The staff concludes that the design of this system satisfies the staff's guidelines on interlocks to prevent overpressure of the primary coolant system during low temperature operation.
3. The staff reviewed the interlock for the SIT isolation valves. The staff's position for interlock systems is in BTP ICSB-4, "Requirements for Motor Operated Valves in the [Emergency Core Cooling System] ECCS Accumulator Lines." The staff concludes that the design of these interlocks satisfies the staff's guidelines on valve interlocks to assure the availability of the safety injection tanks.

The staff concludes that the design bases are consistent with the plant safety analyses and the systems' importance to safety. Further, the staff concludes that the design for those systems is adequate to ensure that the systems meet the functional performance requirements for single failures, redundancy, independence, qualification, and testability.

7.7 Control Systems Not Required for Safety

7.7.1 Discussion

The staff reviewed I&C systems for normal operation that do not perform safety functions after anticipated operational occurrences or accidents but control those plant processes that affect plant safety. The control systems reviewed are:

- reactivity control systems

- pressurizer pressure control system (PPCS)
- pressurizer level control system
- megawatt demand setter
- feedwater control system (FWCS)
- steam bypass control system (SBCS)
- reactor power cutback system
- boron control system
- in-core instrumentation system
- ex-core neutron flux monitoring system
- boron dilution alarm system
- alternate protection system
- process component control system
- cavity flooding system (CFS)
- hydrogen mitigation system (HMS)
- reduced RCS inventory instrumentation
- SGTR detection instrumentation
- advanced control complex (ACC)
- discrete indication and alarm system
- integrated process status overview
- nuclear steam supply system integrity monitoring system
- data processing system

The acceptance criteria for the review of these systems are GDC 13, "I&C," and GDC 19, "Control Room." These I&C systems use digital technology to meet the design requirements and, hence, the staff's concerns and positions in Sections 7.1, 7.2, and 7.3 apply also to these systems.

7.7.1.1 Reactivity Control Systems

Reactivity control systems are implemented in the power control system (PCS). Reactivity is controlled by adjusting the CEAs for rapid reactivity changes and by adjusting the boric acid concentration for slow reactivity changes. The RRS automatically adjusts reactor power and reactor coolant temperature to follow turbine load transients within established limits. The RRS receives a turbine load index signal and reactor coolant temperature signals. These signals are processed to produce a temperature error signal that is then biased by the power range neutron flux to generate a compensated temperature error signal. The CEA rate program uses the temperature error signal to determine CEA position adjustments, which are used by the CEDMCS to adjust CEA positions. The operator may also control the CEDMCS manually from the CEDM operator's module in the MCR.

If the measured average temperature is significantly higher than the reference temperature setpoint calculated from the turbine load index signal, an automatic withdrawal prohibit signal is generated that prohibits the outward movement of the CEAs.

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The CEDMCS monitors CEA positions by counting the number of pulses within each CEDM as the CEDM is raised or lowered. The pulse-counting CEA position signal for each CEA is reset to zero whenever the rod drop contact in the reed-switch housing is closed. This CEA position information appears as a digital display among the control board displays.

The CEDMCS sends position information signals to the DPS for the DPS CEA monitoring displays and alarms and sends signals to the core operating limit supervisory system (COLSS).

The CEDMCS receives a CWP interlock from the PPS based on high pressurizer pressure or a CPC pre-trip signal. This interlock stops the withdrawal of all CEAs. The operator can override the CWP at the CEDMCS operator's module on the MCP.

The CEDMCS receives four CMI interlocks, one from each of the CPCs. The interlocks use a two-out-of-four configuration to stop CEA motion in both directions. These interlocks are initiated when a CEA deviation is detected by either of the two CEACs. The operator can override the interlocks individually.

The CWP and CMI interlocks are linked to the PPS by a fiber-optic data link to ensure separation and independence.

During plant startup and shutdown, and in all cases where the reactor power is below a preset value, the CEAs must be manually controlled. Manual control may be assumed at any other time.

A reactor trip causes the reactor trip switchgear to remove the input motive power from the CEDMCS, releasing all CEAs and allowing gravity to insert them into the core. CEDMCS is thus not required for plant protection.

7.7.1.2 Pressurizer Pressure Control System

The process-CCS includes the PPCS, which controls pressurizer heaters and spray valves to maintain the RCS pressure within specified limits. The pressurizer supplies a water and steam surge volume to minimize pressure variations resulting from density changes in the coolant caused by RCS temperature changes.

A proportional controller uses a pressurizer pressure signal to control the proportional heaters. The heaters increase the pressure by generating additional steam from the liquid volume in the pressurizer. If the pressure continues to decrease, the backup heaters are automatically energized. If the pressurizer pressure reaches a high setpoint, the

heaters are deenergized to ensure the heaters will not further increase pressure. If the pressurizer level decreases to a low-level setpoint, the heaters are deenergized to prevent damage to the heaters. The heaters can also be manually controlled.

The pressurizer pressure signal is also used to control the pressurizer spray valves. The spray valves connect two of the cold legs to the upper head of the pressurizer. The cold-leg water temperature is less than the temperature of the steam in the upper head of the pressurizer. Consequently, the cooler water will condense the steam in the pressurizer upper head, and thereby lower the pressurizer pressure. The spray valves can also be manually controlled.

7.7.1.3 Pressurizer Level Control System

The process-CCS includes the pressurizer level control system (PLCS), which minimizes changes in the RCS inventory by using the charging pumps, the charging flow control valves, and the letdown control valves in the CVCS.

During normal operations, the pressurizer level is programmed as a function of the RCS average temperature to minimize the demands for charging flow and letdown flow. If the level is too high, the PLCS throttles the charging flow to its minimum preset value. If the level is too low, the PLCS increases the charging flow above the letdown flow, thereby increasing the RCS liquid inventory.

The auto-manual control allows the operator to control level manually by controlling the net charging flow rate by moving the charging and letdown flow control valves.

7.7.1.4 Megawatt Demand Setter (MDS)

The PCS includes a MDS, which automatically coordinates turbine generator control with specific plant parameters to prevent exceeding NSSS limits affected by load transients. The operator can perform this supervisory function through the turbine control system (TCS) or, if load is set remotely, manually or automatically through the MDS.

The MDS accepts increases and decreases in power load demands from either the automatic dispatch system or from the local MDS panel. The MDS functions similarly to the systems in Arkansas Nuclear One, Unit 2, and Waterford Unit 3. Power demand is compared with various operating limits including those from the PPS and the COLSS algorithms in the DPS. If the demanded load rate of change exceeds the unit operating limits, an override limit is generated to bring the turbine loading to a level that is consistent with the operating limits.

The MDS calculates overrides based on NSSS trip, balance-of-plant (BOP), and COLSS limits. As a limit is approached, the maximum rate of load increase is reduced to zero, thereby allowing the unit to smoothly attain the power allowed by the unit limiting condition. If an operating limit is exceeded, the MDS will generate a minimum negative rate, and thereby force a load decrease.

Turbine limiting conditions are monitored and used to initiate MDS actions to ensure consistency with TCS operation. These actions include taking the MDS completely out of service for such major upsets as a turbine trip, since fast turbine control actions are required to maintain grid stability during grid frequency disturbances.

7.7.1.5 Feedwater Control System

Feedwater control functions are implemented in the process-CCS. During plant startup and shutdown, a constant-speed startup feedwater pump supplies feedwater flow to the steam generators through feedwater bypass control valves, which are manually controlled to regulate steam generator water level.

The FWCS dedicated to each steam generator automatically controls the steam generator downcomer water level during power operations between 5 percent and 100 percent. Steam generator level is controlled during the following conditions if all other control systems are operating in automatic mode:

- steady-state operations
- 1-percent-per-minute turbine load increases between 5-percent and 15-percent NSSS power and 5-percent-per-minute turbine load increases between 15-percent and 100-percent NSSS power
- 10-percent turbine load steps between 15-percent and 100-percent NSSS power
- loss of one of three operating feedwater pumps
- load rejections of any magnitude

Below 15-percent NSSS power, the FWCS dynamically compensates for the steam generator level signal by sending a flow demand signal to a downcomer valve program where a downcomer feedwater valve demand signal is generated. The signal, or a manual signal from the operator, is passed to the downcomer valve. The signal controls the valve position. When the FWCS is in this control mode, the economizer control valve is closed and the pump speed setpoint is set to near its minimum value.

As NSSS power increases above 15 percent, the downcomer valve closes and the economizer valve opens to regulate the feedwater flow into the steam generator. The steam generator level signal is compensated by the difference between the total feedwater flow rate and the total steam flow rate. The resulting signal is subtracted from the level setpoint signal and sent through a proportional plus integral (PI) controller to produce a total feedwater demand signal. This signal produces an economizer valve demand signal. The operator can also control this signal manually.

The signal is also used to produce a feed-forward demand signal for the feedwater pumps. The pump program generates a pump speed setpoint signal which is passed to one of the feedwater pumps. The operator can also control this signal manually.

The main feedwater system has three 50 percent capacity motor-driven main feedwater pumps normally operating. The FWCS is designed to automatically control steam generator level during a loss of one of three operating feedwater pumps (excluding the startup feedwater pump).

The main feedwater system isolation valves (MFIVs) will close within 5 seconds after receipt of a MSIS, even if the effects of a single failure are imposed. The MSIS is actuated on high containment pressure, high steam generator level, low steam generator pressure, or by manual actuation. The high steam generator level trip feature addresses Unresolved Safety Issue A-47, "Safety Implications of Control Systems."

7.7.1.6 Steam Bypass Control System

The SBCS is implemented in the process-CCS. The SBCS controls the positioning of the turbine bypass valves through which steam is bypassed around the turbine into the condenser. The SBCS increases unit availability by using the turbine bypass capacity to remove excess thermal energy after turbine load rejections by selecting turbine bypass control valves and controlling the release of steam. This avoids unnecessary reactor trips and prevents the opening of the pressurizer or secondary system safety valves.

The reactor power cutback system (RPCS), which is implemented in the PCS, operates with the SBCS to reduce the required capacity of the turbine bypass valves. The SBCS maintains an even load on the reactor as the turbine is brought up to load. When the reactor is heating up or cooling down the SBCS removes excess NSSS energy and controls the rate of RCS temperature changes.

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The valves can be automatically or manually modulated to control the flow of the steam through the valve, and can be quickly opened to prevent rapid NSSS pressure surges caused by heat sink rejections, and can be rapidly closed when the condenser pressure exceeds a set pressure.

7.7.1.7 Reactor Power Cutback System

The RRS, PPCS, PLCS, MDS, FWCS, and SBCS work together to control minor changes in power and flow expected during normal NSSS operation. However, the PCS includes the RPCS to maintain the NSSS within the control band ranges during certain large plant transients such as a large turbine load rejection, turbine trip, or loss of one of three online main feedwater pumps. Under these conditions, the NSSS can be maintained within the control band ranges by reducing NSSS power more rapidly than by a normal high-speed CEA insertion. Rapidly reducing NSSS power will also increase the thermal margin to accommodate inward CEA deviations (including rod drops) without a reactor trip.

The RPCS accommodates certain imbalances by dropping one or more preselected groups of full-strength CEAs into the core to reduce power incrementally. The RPCS also sends control signals to the turbine to rebalance the turbine and reactor powers, and to restore steam generator water level and pressure to control values.

The RPCS receives signals indicating loss of any operating feedwater pump (two signals for each pump), two cutback demand signals from the SBCS, and four cutback demand signals from the CPCs (one signal from each CPC). A two-out-of-two logic actuates the system for load rejections or loss of a feedwater pump. A two-out-of-four logic actuates the system for CEA deviations to be consistent with the two-out-of-four trip initiations from the CPCs and the PPS. The CEDMCS two out of four logic for the CPC cutback demand may be converted to two out of three during CPC maintenance. An operator may also actuate the system manually.

DPS CEA logic selects a predetermined pattern of CEA groups for the RPCS upon receiving signals for NSSS power, CEA positions, and coolant temperatures. The logic informs the RPCS of the CEA groups selected for dropping during a RPC. If the DPS CEA selection logic is inoperable, the RPCS control logic switches to the manual select mode, in which the operator at the RPCS console selects the CEA groups to be dropped.

RPCS actuation initiates the dropping of the preselected pattern of CEAs. The CEDMCS prevents the RPCS from dropping CEA groups that are not intended to drop for a RPC. Other groups of rods are inserted either automati-

cally by the RRS or manually by the operator as necessary. The actuation logic also temporarily changes plant control to a turbine follow mode by rapidly reducing turbine power to 60-percent power and further reducing power if necessary to balance turbine power with reactor power.

7.7.1.8 Boron Control System

The process-CCS gives the operator information for regulating and monitoring boron concentration in the reactor coolant. The CVCS enables the operator to control the dilution and addition of boron. Boric acid stored in the volume control tanks can be added to the reactor coolant.

To assist the operator in maintaining the proper boric acid concentration in the RCS, indications of boron in parts per million (ppm) are available on a DIAS discrete indicator and the DPS displays. The operator can also use the DPS to observe trends of the concentration. These signals are supplied by the boronometer. Additional discrete indicators and CRT displays indicate reactor makeup water flow and boric acid makeup flow, which can be used to determine whether boron is being added or diluted.

7.7.1.9 In-Core Instrumentation System

The 305 in-core self-powered rhodium neutron flux instruments monitor the core power distribution locations during operation. The design includes no traversing in-core probes. These detectors are distributed so that the axial power shape and the radial power shape can be readily determined. The signals from the detectors are sent to the DPS for monitoring and display. The DPS compensates for background radiation, beta-decay delay, and rhodium depletion using digital signal-processing routines.

The in-core instrumentation system determines the gross power distribution in the core for various operating conditions and sends the DPS data for estimating fuel burnup in each fuel assembly and for evaluating thermal margins in the core.

The in-core detectors can send azimuthal and axial power distribution information for calibrating the ex-core detectors. The safety-related ex-core neutron flux monitoring system indicates the flux power and axial distribution for the RPS.

7.7.1.10 Ex-Core Neutron Flux Monitoring System

The ex-core neutron flux monitoring system includes neutron detectors located around the core and signal-conditioning equipment located in the control room area.

Ex-core detector channels give the operator source level neutron flux information during extended shutdown periods, initial reactor startup, startups after extended shutdown periods, and following reactor refueling operations. Each channel consists of two sections having multiple BF_3 proportional counters; a preamplifier outside the reactor shield; and a signal-processing drawer containing power supplies, a logarithmic amplifier, and test circuitry. High-voltage power to the proportional counters is terminated when the neutron flux is several decades above the source level to extend the detector life. These channels send information for display and audio count-rate but do not perform control or protective functions. Ex-core detector channels also send the RRS flux information in the 1-percent to 125-percent power operating range for automatic turbine load-following operations.

7.7.1.11 Boron Dilution Alarm System

The concentration of soluble boron in the RCS affects the control of reactivity in the reactor core. The BDAL receives signals from the ex-core detectors to detect boron dilution events while the reactor is in Modes 3 to 6. The DIAS and DPS monitor the BDAL to ensure detection and alarming of the event.

The alarm setpoint is periodically automatically lowered to a fixed amount above the current neutron flux signal. The alarm setpoint will only follow decreasing or steady flux levels, not an increasing signal. The DIAS and DPS CRTs display the current neutron flux indication and alarm setpoint. A reset capability enables the operator to acknowledge the alarm and initialize the system.

7.7.1.12 Alternate Protection System

APS functions are implemented as part of the process-CCS. The APS augments the RPS to address 10 CFR 50.62 requirements for reducing the risk of anticipated transients without scram (ATWS) and the use of ATWS mitigating systems actuation circuitry (AMSAC).

The APS design includes an alternate reactor trip signal (ARTS) and AFAS that are separate and diverse from the PPS. The ARTS equipment is a diverse means to decrease the possibility of an ATWS, and the AFAS provides added assurance that a source of safety-related feedwater will be available following an ATWS.

The ARTS will initiate a reactor trip when the pressurizer pressure exceeds a nominal setpoint of 16.7 MPa (2420 psia). Turbine-trip signals can also initiate ARTS if the RPCS is not available. The ARTS circuitry is diverse from that of the RPS. The ARTS design uses a two out of two logic to open the CEDM motor generator output

contactors, thus removing motive power to the RTSS, thereby causing the CEAs to drop into the core.

The AFAS will initiate EFW when the level in either steam generator decreases below 23.4 percent of the 0-1016 cm (0-400 in.) steam generator wide-range level. The EFW components are actuated by sending isolated AFAS signals to the CCL in the ESF-CCS.

The DIAS receives the ARTS and AFAS trip status, pressurizer pressure, and steam generator 1 and 2 level parameters for display. The DPS receives the same data as provided to DIAS.

The APS design requirements in 10 CFR 50.62, "Requirements for Reduction of Risk From ATWS for Light-Water-Cooled Nuclear Power Plants," state that each pressurized-water reactor must have equipment, from the sensor output to the final actuation device, that is diverse from the RTS, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under ATWS conditions. This equipment must perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the RTS. In its original submittal, ABB-CE stated that the EFW actuation circuit design uses the CCL module in the ESF-CCS to satisfy the ATWS rule. ABB-CE did not include sufficient information for the staff to verify that this module is diverse from those in the RTS (DSER Open Item 7.7.1.12-1).

ABB-CE clarified the issue regarding the diversity of the EFW actuation circuitry by stating that the protective system actuation of the EFWS uses circuitry in the PPS and a component control module in the process-CCS. The process-CCS is diverse and independent from the PPS and the ESF-CCS. Initiation signals generated independently by the two systems are logically combined in hardware downstream of the two systems, using "hardwired OR" circuits, so that either system can actuate the EFWS. This hardware is located at the final actuation devices of the EFWS (i.e., the switchgear controlling the EFWS pumps and valves). Isolation is provided at the process-CCS signal to the "hardwired or" to maintain electrical isolation from the ESF-CCS. This clarification regarding diversity between the EFWS actuation modules in the PPS and the process-CCS resolved DSER Open Item 7.7.1.12-1.

In the original submittal, the staff could not confirm a diverse turbine trip capability. ABB-CE did not submit a detailed design of a diverse turbine trip capability in the ATWS actuation circuits (DSER Open Item 7.7.1.12-2). ABB-CE later stated the APS uses equipment from the sensor output to the final actuated device that is diverse from the RTS to automatically initiate a turbine trip under ATWS conditions. The AMSAC turbine trips are initiated

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by undervoltage relays in the CEDMCS. Since this trip function is not part of the PPS, the diverse turbine trip capability requirement in 10 CFR 50.62(c) is satisfied. This resolves DSER Open Item 7.7.1.12-2.

In 10 CFR 50.62, each pressurized-water reactor (PWR) designed by Combustion Engineering or Babcock and Wilcox must have a diverse scram system from sensor output to interruption of power to the control rods. This scram system must perform its function in a reliable manner and be independent of the RTS (from sensor output to interruption of power to the control rods). The System 80+ ARTS is consistent with these requirements.

7.7.1.13 Process Component Control System (Process-CCS)

The process-CCS controls non-safety-related pumps, valves, heaters, and fans and other components and sends process variables and CCS status information to the DIAS and DPS for plant monitoring.

The system permits component assignments to independent non-Class 1E subgroup segments to minimize the plant impact caused by component or system-level failures. Standard CCL and I/O links allow for the various types of components to be controlled, as described in Section 7.3. The design includes SCL to supervise subgroups of components and to generate system status information for the DIAS and DPS. The design also includes master transfer capability with isolation to disable all MCR controls and enable component controls for the RSR.

The process-CCS Group Controllers include a sequencer to automatically start and load the AAC source with essential non-safety loads during LOOP events coincident with a loss of non-safety onsite power. When an EDG is out of service, this sequencer is blocked, permitting the ESF-CCS sequencer to automatically load selected Class 1E division loads.

The process-CCS accommodates both local and remote distribution of I/O multiplexers. The system architecture uses multiple redundant CCL processors with redundant internal data communications.

Both the ESF-CCS (described in Section 7.3) and the process-CCS are microprocessor-based systems with programmable logic for their unique component control applications. Diversity between the two designs is achieved through the use of diverse microprocessors, software, and communication networks to offer a defense-in-depth approach for ensuring maximum availability of normal and emergency systems. The staff concludes that this level of diversity is acceptable since the likelihood of

potential common-mode or common-case failures of both the systems that can lead to the loss of the required function is small. Section 7.2 contains the staff review and acceptance of System 80+ I&C diversity.

7.7.1.14 Parameters Sensed by Control and Monitoring Systems

A control system may use redundant independent Class 1E sensors from each channel of the protection system. For each sensed parameter, the control system monitors all four redundant instrument channels through fiber-optic links to ensure electrical independence. Signal validation logic in the control and monitoring systems detects by-passed or failed sensors, thereby ensuring that these signals do not cause erroneous control actions. The validation logic also selects the sensed value to be used in the control system.

To ensure correct performance of the validation logic, it will be designed in accord with the same quality assurance program as for Class 1E systems. The logic will be implemented within the respective control systems. The fiber-optic links transmitting signals from the Class 1E protection channels to the control systems ensure that no electrical failures can adversely propagate from the control systems to the protection system. This isolation logic ensures that an electrical fault on any single sensor will not cause the signal validation logic to fail. The sensors are qualified to operate in adverse seismic and environmental conditions. Since the sensors are not affected by such conditions, the protection system will not be impaired.

To ensure that no undetected failures exist within the signal validation logic, three levels of testing are performed:

1. The hardware in which the logic is implemented is continuously tested to detect electronic component failures.
2. The validation logic continuously transmits the controlling signal to the DPS from the control system while the raw sensor data is transmitted to the DPS from the protection system. The DPS calculates its own valid signal and continuously compares it to the valid signal from the control system. Unacceptable deviations are alarmed.
3. Sensor signals to the control system may be manually disturbed to verify that the logic correctly discriminates the bad data.

7.7.1.15 Cavity Flooding System

The CFS directs flow from the IRWST to flood the reactor cavity during a severe accident. The CFS is controlled

manually from the control room. CFS controls and instruments include independent indications powered from vital power buses A and B, and independent controllers powered from the corresponding vital power buses.

7.7.1.16 Hydrogen Mitigation System

The HMS allows adiabatic, controlled burning of hydrogen at low concentrations during degraded-core accident conditions. HMS ignitors in separate channels are manually actuated from the control room. HMS controls and instrumentation are discussed in Section 6.2.5. Electrical power distribution is discussed in Section 8.3.

7.7.1.17 Advanced Control Complex

The Nuplex 80+ ACC design integrates I&C systems for the NSSS and BOP systems. The ACC comprises the MCR, computer room, RSR, the TSC and visitors gallery, the I&C equipment rooms, CEDMCS and non-1E power and equipment rooms, and 1E power and equipment rooms.

The ACC is designed with consideration of exposure to fires that could result in damage and require MCR personnel evacuation. Redundant channels of Class 1E equipment are located in different equipment rooms that have separate and independent power supplies, HVAC, and 3-hour fire barriers. Master transfer and isolation of controls and indications are provided for equipment in the MCR and the RSR. The two control rooms are electrically isolated from each other so that a fault in one room cannot propagate to the other room.

The ACC Class 1E equipment is seismically qualified as seismic Category I, in accordance with IEEE 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The equipment is to be qualified for Environmental Category J conditions of 23-26 °C (73-78 °F) continuous temperature at atmospheric pressure with relative humidity of 20 to 60 percent continuous and 10 Gy (10^3 rad) gamma integrated dose. Environmental Category J encompasses normal and DBA conditions for the control room. The environmental qualification for other control building areas is to be 29 °C (85 °F) continuous temperature at atmospheric pressure with relative humidity of 20 to 100 percent continuous and 10 Gy (10^3 rad) gamma integrated dose.

In the original submittal, ABB-CE had not given the staff sufficient information to evaluate provisions for monitoring the environmental conditions (e.g., temperature, moisture, humidity, chemical pollutants) and conditions that could cause functional degradation (e.g., pipe breaks, fires, loss

of ventilation, spurious operation of fire protection systems) within the equipment. Operating experience indicates that such monitors are necessary for alerting plant personnel to equipment operating in an environment beyond its design limits. This was designated DSER Open Item 7.7.1.17-1.

ABB-CE later stated that the HVAC and fire protection systems monitor the environmental space around the equipment cabinets. The HVAC system and the fire protection systems are discussed in Sections 9.4 and 9.5, respectively. DSER Open Item 7.7.1.17-1 has been resolved as part of the staff's acceptance of the HVAC design.

The MCPs are compact workstations that integrate miniature backlit component control switches, process controllers, discrete indicators, alarm tiles, message windows, and video display units (CRT, plasma, and electro-luminescent displays) so that both safety-related and non-safety-related display devices are routinely used by the operator.

The MCPs maintain structural integrity so that no control room missile hazards result as a consequence of a seismic event. Any safety-related Class 1E components mounted in the panels are seismically qualified to perform their safety functions.

All NSSS and BOP instruments, controls, and alarms link to the DIAS, DPS, or CCS for routing to the control panels; except for operators' modules dedicated to specific plant components (e.g., PPS, TCS, CEDMCS).

To minimize the possibility of damage to multiple channels within the MCPs or RSPs, the panels employ low-energy circuits (less than 50 V) to the maximum extent possible, fire-retardant materials and smoke detectors, and electrically independent channels of circuits.

If multiple redundant channels are damaged, the MCR circuits are fault isolated from the electronic components with which they interface. All MCP and RSP circuits are passive. Momentary contacts are used for all switches with the memory of control panel commands retained only in the electronic circuits in the I&C equipment rooms. The MCR, RSP, and the I&C equipment rooms are located in separate fire zones.

Control to the RSR is transferred in two steps. If a fire is detected within an MCR panel section, power is removed from the affected panel section by activating disconnect switches for each channel located within the MCR. This transfer removes power from the affected panel components, preventing the transfer of erroneous operator commands to the system electronic circuits in the I&C

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equipment rooms. Control is then transferred to the RSR by activating control transfer switches for each channel located at the exits of the MCR or in the I&C equipment rooms.

The DPS includes links with the TSC, EOF, and nuclear data link as needed to conform to NUREG-0696, "Functional Criteria for Emergency Response Facilities."

7.7.1.18 Discrete Indication and Alarm System

The DIAS displays and alarms include signal validation, automatic ranging, alarm filtering, alarm prioritization, pattern recognition, and other features to enhance the man-machine interface. The DIAS receives analog and digital signals from both safety-related and non-safety-related systems, analyzes the data, and presents the information to the operator on discrete indicators, alarm tiles, and message windows located on the MCPs. The DIAS and DPS integrate alarm and process status information in the panel displays. DIAS supports continuous plant operation if the DPS becomes unavailable.

The DIAS is a segmented, distributed architecture. The system consists of a DIAS-P segment for display of postaccident RG 1.97 variables and DIAS channels N1 through N7 for the remaining parameters. Each segment consists of I/O data links and multiplexers, CPUs, and display and alarm devices.

The segmented DIAS includes independent hardware and fault resistance. The DIAS-P is physically independent from the remaining DIAS-N segments and the DPS, so that a single failure will not cause a loss of more than one of the three display methods (DIAS-P, DIAS-N, DPS). The redundant I/O data links and CPUs in each segment allow for transfer to the backup CPU without interrupting the information being displayed on the control panel devices.

Fiber-optic data links provide isolation between the redundant safety-related channel I/O and DIAS CPUs, and between the DIAS CPUs, the MCP I/O multiplexers, and the RSP I/O multiplexers.

The data from both safety-related and non-safety-related sources is scanned at a rate that accommodates the requirements for alarm checking, signal conversion, and signal validation. The DIAS receives signals from the ESF-CCS, the process-CCS, the PCS, the RTSS, ex-core and in-core nuclear instrumentation, the CPCs, the NSSS integrity monitoring system, the PPS, electrical systems components, HJTCs and CETs, and the motor-generator sets.

Input data, calculated values, or parameters for another DIAS segment are available through a data network that

interconnects each of the DIAS-N segments. This permits sharing of all available plant data needed for the signal validation and individual alarm logic algorithms. To maintain independence, the DIAS-P is not part of this data network.

Redundant CPUs and data links permit on line maintenance, testing, and repair without affecting the amount and quality of information displayed for the operator. If the redundant CPUs and data links in the segment fail, the diverse and independent DPS will control its alarms and displays with its own data.

The DIAS CPUs use dynamic complementary metal-oxide semiconductor random access memory so that the software for all tasks resides in memory and does not depend upon mechanical rotating storage devices to function. The vital instrument power buses connect to backup batteries to prevent loss of memory caused by power interruptions and to automatically restart the system without operator intervention.

Power is distributed to each segment so that a loss of one division of Class 1E vital instrument bus power will affect only one of the redundant CPUs, associated I/O multiplexers, or a set of control panel displays.

The DIAS processors for each segment control discrete indicators on the main and remote shutdown control panels. The discrete indicators are flat-panel display devices with touch screen controls, and allow the operator to access additional information that is normally not presented on the upper-level displays.

DIAS processors control the discrete indicators for each segment. The DIAS processor receives signals from safety-related and non-safety-related instrument channels and performs a validation and range selection process, which includes comparing the signal to those from the postaccident monitoring channel(s). The DIAS processor sends the output signal to the appropriate discrete indicator device. Selected parameters are displayed continuously with alternate parameters available for display if selected at the discrete indicator.

Each discrete indicator has a CRT access selector to retrieve appropriate menus for related detailed data on CRT displays. The discrete indicator sends the menu selection to the DPS through the DIAS-DPS data link, which transmits all validated parameter values from the DIAS to the DPS for display on the CRTs. The DPS independently performs the validation algorithm and periodically checks its results against the DIAS results. If the DPS finds a discrepancy in the validated parameter, it actuates a discrepancy alarm. The failure of an individual

discrete indicator does not affect the DIAS CPU, which will continue validating data and transmitting validated data to the DPS and IPSO for display.

If the DIAS logic cannot validate a parameter, it informs the operator by means of a discrete indicator, so that the parameter of choice can be selected and displayed. If DIAS-N parameter validation is inconsistent with the DPS validation, the operator will use the DIAS-P display for monitoring and if necessary, based on the DIAS-P indications, take appropriate safety actions with the hardwired control switches.

The DIAS-P and various DIAS-N segments each contain duplicates of the algorithms for monitoring ICC. These same algorithms are also duplicated in various DIAS-N segments. The DIAS-P indicators will continuously display alphanumeric values for all RG 1.97 Category 1 parameters. The parameters selected by ABB-CE for monitoring postaccident conditions conform in type and range with those recommended in RG 1.97.

Individual DIAS segments are designed so that a failure of one segment's processor or a communications link will not affect any other segment's alarms. Although additional failures may result in the degradation of one (or more) segment's displays, the DPS CRT display and printer log will continue to independently alert the operator to alarms and alarm discrepancies and display descriptive alarm information.

Class 1E instrument channels are seismically and environmentally qualified up to and including the channel isolation device (fiber-optic modems) so that the instrument channel is not degraded. This is consistent with guidance in Table 1 of RG 1.97.

The DIAS displays and CPUs are designed as non-Class 1E to meet the requirements for control room and electronic equipment room ambient temperature, pressure, and humidity [29 °C (85 °F), 20- to 100-percent humidity at atmospheric pressure]. All DIAS displays and CPUs are seismically qualified for physical and functional integrity to ensure information will be available in the control room.

The DIAS does not perform a safety function since it only monitors and displays data; however, it is designed in accordance with a quality program commensurate with the intended use of the equipment. All DIAS I/O equipment is qualified to avoid generating faults that would degrade Class 1E signals from Channels A, B, C, and D. The qualified Class 1E fiber-optic data links maintain isolation.

7.7.1.19 Integrated Process Status Overview

The integrated process status overview (IPSO) displays DPS and DIAS-N information at a single location that the operators and supervisory personnel require for quickly assessing overall plant status. The IPSO is a large top-level DPS CRT display above the master control console in the MCR. The display indicates Priority 1 alarms calculated by the various DIAS-N segments, deviations from control setpoints, key parameter values, and system operational status and nonoperational availability in a schematic representation. The IPSO displays the major heat transport path systems and systems that are required to support the major heat transport process, either power or safety related. These systems include those that must be monitored for availability according to RG 1.47.

The IPSO is a redundant system powered by battery-backed non-vital instrument buses. If the IPSO fails, the operator can monitor the DIAS display to assess plant conditions.

The IPSO will maintain physical integrity during seismic events.

7.7.1.20 NSSS Integrity Monitoring System

The NSSS integrity monitoring system detects selected conditions that indicate a deterioration or that could lead to a deterioration of the RCS pressure boundary. The system consists of the internals vibration monitoring system (IVMS), the acoustic leak monitoring system (ALMS), and the loose parts monitoring system (LPMS).

The IVMS supplies data from which changes in the motion of the reactor internals can be detected. The IVMS also supplies data that can be used to diagnose the reasons for these changes. The IVMS uses time variations in the neutron flux measured by the ex-core neutron detectors, which are caused by changes in the neutron path lengths caused by motion of the reactor internals.

The ALMS detects leaks at specific locations or within specific components in the primary system. The ALMS is designed to meet relevant guidance of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." The ALMS provides one method of determining the position (closed or not closed) of the pressurizer safety valves as defined by RG 1.97. The ALMS uses piezoelectric accelerometers attached to the components for which the leak is to be detected. Leakage of a fluid results in turbulent fluctuations in pressure, sending stress waves through the component, which are detected by the accelerometers. Signal conditioning results in an indication of a possible leak at that component.

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The LPMS detects loose parts within the primary pressure boundary and supplies diagnostic information to assist in determining (1) the nature of the loose part (fixed or free), (2) the location of the loose part, and (3) the characteristics of the loose part (size, mass, velocity). The system is designed to meet the guidance of RG 1.133, "Loose Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

The LPMS uses accelerometers attached to the components for which the loose part is to be detected. The impact of a loose part on the boundary surface of a system sends a series of waves through this boundary, which are detected by the accelerometers. Signal conditioning results in an indication of the size, mass, velocity, and location of the part.

7.7.1.21 Data Processing System

The DPS is a computer-based system that gives the operator plant status information from plant sensors, other I&C systems, and self-contained algorithms in application programs. The DPS provides this information on a real-time basis and as historical data (such as for trends).

The DPS acquires data through data links from the other plant systems; validates sensed parameters; executes NSSS application programs and BOP performance calculations; monitors plant safety and general status; presents safety parameter status, and calculation results for CRT display in the control room, TSC, and EOF; maintains logs; and determines alarm conditions.

The DPS comprises host processors, intelligent display processors, CRTs, and other support devices. The host processors use software applications to process the received data and transmit computed results to the intelligent display system. The intelligent display system processes the data for display on the CRTs, and processes operator requests made on the CRT touch screens.

Through host processor, peripheral redundancy, and a distributed design, the DPS accommodates the failure of any single hardware element so that no single failure within the DPS will disable any of the DPS functions. DPS data links acquire plant process data from other plant systems and transmit it to the host processors. The data links between the host processor and plant I&C systems are fiber-optic for isolation.

Each host processor has a system console for the programmer and consists of a dual CPU. One CPU is dedicated to I/O and demand tasks, the other CPU is dedicated to periodic tasks.

High-speed line printers record information for the programmer, control room operating staff, TSC, and EOF.

The DPS control room operator primarily uses touch screen color CRT workstations and other touch panel devices, such as annunciator tiles.

All applications are programmed using structured programming rules and techniques. The software comprises modular, structured programs. DPS online operation minimizes reliance on any electromechanical peripherals. All major applications are memory resident and are structured to allow continued execution in the event of a disk, printer, or magnetic tape failure. ABB-CE states that these critical function monitor software modules are verified and validated in accordance with Nuclear Safety Analysis Center 39, "Verification and Validation for Safety Parameter Display Systems." ABB-CE responded to the staff's RAI on software V&V (DSER Open Item 7.7.1.21-1) by submitting a SDP. The staff's acceptance of the SDP, as discussed in Section 7.1.4, resolved DSER Open Item 7.7.1.21-1.

The DPS NSSS application programs give the operator information and alarms to assist in maintaining the plant within specified limits and to evaluate the performance of the reactor core. These applications are:

- core operating limit supervisory system
- CEA position monitoring
- CEA PDIL/PPDIL monitoring
- CEA out-of-sequence monitoring
- CEA deviation monitoring
- CEA trip program
- CEA reassignment
- CEA exposure accumulation
- in-core detector processing
- xenon reactivity prediction
- reactivity balance
- CPC deviation monitoring
- CEAC deviation monitoring
- PPS deviation monitoring
- critical function monitoring
- time-dependent variables processing
- reactor power cutback CEA selection
- data snapshot collection program
- historical data storage and retrieval
- sequence of events
- ESF computer-aided test program

The DPS contains the following application programs for evaluating BOP performance:

- feedwater heater performance calculations
- condensate pump performance calculations

- steam generator performance calculations
- unit generation calculations
- turbine performance calculations
- moisture separator reheater performance calculations

The DPS contains the CFM application program, which includes the critical plant power production function and safety functions. The DPS integrates crucial power production monitoring with TMI Action Plan requirements for ICC monitoring and SPDSs. The algorithm also monitors the success path for safety system critical functions.

The CFM application program displays the status of the following critical functions:

- core reactivity control
- core heat removal
- reactor coolant system inventory control
- reactor coolant system pressure control
- reactor coolant system heat removal control
- containment environment control
- containment isolation control
- radiological emission control
- steam and feed systems control
- vital auxiliary systems control
- electric generation control
- ultimate heat rejection control

The success path monitoring (SPM) program continuously monitors the status and performance of the plant systems and components to determine their ability to satisfy the first eight of the critical safety functions listed above.

The SPM program displays the success path status for each critical safety function and initiates alarms when they become bypassed or inoperable.

7.7.2 Conclusions

The plant control systems and equipment will be highly reliable during steady-state operation and anticipated transient conditions. The reactor protection system FMEA discussed in Section 7.2 encompasses the failure modes of these control systems and demonstrates that these systems are not required for safety. The safety analyses in SAR Chapter 15 do not require these systems to continue functioning after an accident.

The staff reviewed the control systems used for normal operation that are not relied upon to perform safety functions but that control plant processes having a significant effect on plant safety. These control systems include the reactivity control systems and the control systems for the primary and secondary coolant systems.

The functional design of the control systems is acceptable and meets the relevant requirements of GDC 13 and 19. This conclusion is based on the staff's review of the plant transient response to normal load changes and anticipated operational occurrences such as reactor trip, turbine trip, and upsets in the feedwater and SBCSS. The control system functional designs can maintain system variables within operating limits. This meets the requirement of GDC 13.

ABB-CE stated that the APS uses equipment from the sensor output to the final actuated device that is diverse from the RTS to automatically initiate a turbine trip under ATWS conditions. The AMSAC turbine trips are initiated in undervoltage relays of the CEDMCS. Since this trip function is not part of the PPS, the diverse turbine trip capability requirement in 10 CFR 50.62(c) is satisfied.

The staff reviewed features of the control systems for both manual and automatic control of the process systems to maintain plant variables within prescribed operating limits. The control systems permit actions to operate the plant safely during normal operation, including anticipated operational occurrences; therefore, the control system functional designs satisfy the requirements for normal plant operation in GDC 19, "Control Room."

The conclusion of the analysis of anticipated operational occurrences and accidents as presented in SAR Chapter 15 have been used to confirm that plant safety is not dependent upon the response of the control systems. The staff confirmed that failure of the systems themselves or as a consequence of supporting systems failures, such as power sources, does not result in plant conditions more severe than those bounded by the analysis of anticipated operational occurrences.

The staff confirmed that the effect of anticipated operational occurrences and accidents does not result in control system failures that would cause plant conditions more severe than those bounded by the analysis of the events, and the staff finds that the control system functional designs are not relied upon to ensure plant safety.

The staff finds that the EFW actuation circuit design uses CCL modules that are diverse from the ESF-CCS to satisfy the ATWS rule (10 CFR 50.62). The staff finds that the ARTS is designed to perform its function independent of the RTS. In a subsequent submittal, ABB-CE corrected the design drawing, thereby resolving DSER Open Items 7.7.1.12-1 and 7.7.1.12-2.

ABB-CE gave the staff sufficient information to verify that the design includes appropriate provisions for monitoring the environmental conditions (e.g., temperature, moisture,

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humidity, chemical pollutants) and conditions such as pipe breaks, fires, loss of ventilation, and spurious operation of fire protection systems that could degrade functional capability of the equipment. Operating experience indicates that such monitors are necessary to alert plant personnel when the equipment is operating in an environment beyond its design limits.

The reliable functioning of the PPS and the non-safety-related systems depends upon the reliable operation of the multiplexers in the diverse digital networks. ABB-CE addressed multiplexer vulnerabilities by (1) evaluating high-quality multiplexers that meet the requirements of industrial standards, (2) assessing vulnerabilities to CMFs, and (3) using hardwired essential systems controls that bypass the multiplexer networks.

In the original submittal, ABB-CE stated that the DPS provides non Class 1E redundancy for the DIAS. ABB-CE did not describe the data communications network protocols to be used for the diverse data communication networks. ABB-CE is considering using ArcNet for the DIAS network, and EtherNet for the DPS network. EtherNet local area networks are vulnerable to failure when the data rate on the network approaches the network limit. If the DIAS network fails, increased traffic on the DPS network may cause a complete loss of indications and alarms. The staff required that ABB-CE address the performance criteria and the testing methodologies for the data network designs. ABB-CE must ensure that the data networks can perform reliably during high-traffic periods of operation. These issues were designated DSER Open Item 7.7.3-1.

ABB-CE later stated that the DIAS network communicates data from the data acquisition processors to the display devices at a constant rate. All of the data from the data acquisition nodes is communicated all of the time and always at the same frequency. A change in plant state will change the values communicated for some parameters, but will not change the number of parameter values communicated in a communication cycle. The arrangement of information on a display screen is stored in the display device, and therefore, does not need to be communicated over the network when the display selection is changed. Thus, a change in plant state or operator activity at the display devices does not change the data communication load on the DIAS network.

The DIAS is designed in such a way that the desired data communication uses only 50 percent of the capacity of the network. The capacity is determined by analysis and confirmed by test. Since the communication load is constant, and well within the network capacity, adequate response time performance can be assured. Performance

will be tested to ensure that actual communication loads are within bounds, such that actual response time performance satisfies the design criteria.

The principal communication load on the DPS network is data sent for updating dynamic parameters on display screens. The arrangement of information on a display screen is stored in the display device, and therefore, does not need to be communicated over the network when the display selection is changed. When a display is selected, the display device communicates its data needs to the data acquisition processors. The data acquisition processors then send updated values for the requested parameters with every communication cycle, whether or not the parameter values have changed. Thus, the network communication load does not depend on the plant condition, since the same number of parameters is communicated every cycle. The network load depends upon the total number of different parameters which the data acquisition processors are called upon to put on the network to update the active displays. The DPS is designed to provide a 1-second update of dynamic data under worst case loads, based on the maximum number of display devices and the most demanding resultant total for different dynamic parameters to be updated.

The capability to provide for historical data trends on the DPS may call for large data file transfers from one or more of the data acquisition processors. A test will be performed to verify the network response time meets the update criteria for the worst-case file transfer scenarios during maximum data communication loads.

The DIAS and DPS data networks are electrically separated, independent, and diverse from each other. Since the DPS normally provides displays for the complete plant data base, which includes a redundant display of DIAS information, a failure of the DIAS would not increase the communication load of the DPS network.

The staff finds ABB-CE's description and the design of the data networks acceptable because the design of the diverse data systems reduces the likelihood of loss of both systems at the same time. This resolves DSER Open Item 7.7.3-1 regarding network protocols, performance and testing methodologies.

8 ELECTRIC POWER SYSTEMS

8.1 Introduction

In Section 8.1, Table 8-1 of the Standard Review Plan (SRP) (Revision 3, July 1983), the staff states the primary acceptance criteria for evaluating the adequacy of a nuclear power plant's electric power systems. Among these acceptance criteria and guidelines are the applicable general design criteria (GDC) and guidelines of branch technical positions (BTPs), regulatory guides (RGs), and staff reports. The staff issued RG 1.153, "Criteria for Power, Instrumentation, and Control Portions of Safety Systems," and RG 1.155, "Station Blackout," after it issued Revision 3 to the SRP, and the staff used the guidelines from these regulatory guides as the bases for evaluating the adequacy of the System 80+ electric power system design. In addition, the staff proposed applicable regulations for electrical distribution, as follows:

- (1) at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such manner that the offsite source can power the safety buses upon failure of any non-safety bus.
- (2) an alternate power source that is provided to a sufficient string of non-safety loads so that forced circulation could be maintained, and the operator has available a complement of non-safety equipment that would most facilitate the ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip.

The staff has determined that conformance to the applicable GDC and guidelines cited above is a sufficient basis for accepting the electric power systems for the System 80+ design.

ABB-CE referenced more recent versions of the Institute of Electrical and Electronics Engineers (IEEE) standards that have not been endorsed by the Commission. In the draft safety evaluation report (DSER) of September 1992, the staff stated that if it had not formally endorsed the newer version of an IEEE standard being referenced by ABB-CE, the differences between the newer version and the version endorsed by the Commission must be identified, justified, and approved for use on the System 80+ design to ensure that design criteria used are as conservative as those in the standards currently approved by the staff. Industry codes and standards is an NRC policy issue (SECY-91-078, Section II.B of Enclosure 1 to SECY-93-087).

In SECY-91-078, the staff recommended that the Commission approve the position, consistent with past practice, that it will review both evolutionary and passive plant

design applications using the latest codes and standards that the NRC has endorsed. The staff will review unapproved revisions to codes and standards on a case-by-case basis. In its August 15, 1991, staff requirements memorandum (SRM), the Commission approved the staff's position. This was designated as DSER Open Item 8.1-1.

In a subsequent submittal, ABB-CE described differences between staff-approved IEEE standards and the current version of those IEEE standards which have not been endorsed by an NRC regulatory guide. However, for some IEEE standards, ABB-CE did not clarify which version of an IEEE standard that the System 80+ design complies with. In a meeting held on February 9, 1993, the staff asked ABB-CE to specify the changes between the old and current version of IEEE standards that are deemed applicable for the System 80+ design. By letter dated June 11, 1993, ABB-CE submitted that information. The staff reviewed this information and concludes that ABB-CE has noted the differences between the old and updated IEEE standards which the NRC has not formally endorsed, has justified the differences, and has addressed the specific requirements of those standards that are committed for the design. The design criteria of the updated IEEE standards are as conservative as those standards formally endorsed by the staff. On this basis, DSER Open Item 8.1-1 is resolved. The requirements of the NRC policy issue on industry codes and standards are satisfied.

In the sections that follow, the staff presents its evaluation of the System 80+ offsite electric power system design, the onsite electric power system design, and the manner in which these designs satisfy the acceptance criteria cited above.

8.2 Offsite Power System

The offsite power system is the preferred source of power for the System 80+ design. This system comprises the grid, transmission lines, transformers, switchgear components, and associated control systems that supply electric power to safety-related equipment and other equipment. The electrical grid supplies energy for the offsite power system. The basis for the design of the offsite power system (assuming that the onsite power systems are not available) is to supply sufficient capacity and capability to ensure that the specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded and to ensure that core cooling, containment integrity, and other vital functions will be maintained in the event of postulated accidents. The staff's objective is to determine that the design of the offsite power system satisfies the requirements of GDC 17, "Electric Power Systems," and GDC 18, "Inspection and Testing of Electric Power System," and that the system

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will perform its design functions during normal plant operations, anticipated operational occurrences, and accident conditions.

8.2.1 Offsite Power System Outside the System 80+ Scope of Supply

Two separate and independent transmission lines from the transmission system terminate in the two different switchyards (preferred switchyard Interface I and preferred switchyard Interface II). Separate and independent overhead lines will connect the main and reserve auxiliary transformers of the System 80+ design to the switchyards. The portion of the two separate and independent circuits from the utility grid system to the high terminal of the main transformers and reserve transformers is not within the scope of the System 80+ design. GDC 17 requires that electric power from the transmission network to the on-site distribution system be supplied by two physically independent circuits and GDC 18 requires that electric power systems important to safety be designed to permit appropriate periodic inspection and testing to satisfy these requirements, ABB-CE noted the following requirements for offsite transmission lines for combined license (COL) applicants who reference the System 80+ design:

- (1) The design shall include at least two separate and physically independent circuits from the transmission network, each capable of supplying the plant's necessary safety loads and other equipment.
 - (2) The grid that supplies offsite power shall remain stable for (a) a sudden loss of the largest single supply to the grid (b) a sudden loss of the largest load or a major load center, and (c) the outage of the most critical transmission line, including the sudden loss of all lines on a common right-of-way, the sudden loss of a substation, and the delayed clearing of a three-phase fault at any point on the system because a breaker failed to open.
 - (3) The COL applicant shall analyze the stability of the grid. This analysis shall demonstrate that the 13.8-kilovolt (kV) non-safety buses do not subject the reactor coolant pumps (RCPs) to sustained frequency decays in excess of 3 Hertz per second (Hz/sec).
 - (4) To satisfy the flow conditions for the reactor coolant system (RCS) assumed in the CESSAR-DC Chapter 15 safety analyses, there shall be at least a 3-second time delay between a turbine trip and the loss of offsite power (LOOP).
- In the DSER, the staff requested that ABB-CE add to the CESSAR-DC the following requirements regarding Item 1 (above) for that portion of the offsite power circuits that are outside the scope of design certification:
- (a) The two designated offsite power transmission lines shall be physically separate and independent. The COL applicant should design and route these lines to minimize, to the extent practical, the likelihood of their simultaneous failure under postulated accidents. No other lines should cross above these two circuits. The COL applicant should route these circuits to ensure that no single event, such as a tower falling or a line breaking, can simultaneously affect both circuits in such a way that neither can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded.
 - (b) The two designated offsite power circuits shall terminate at two switchyards that are physically separate and electrically independent to the extent practicable. The COL applicant should specify which source (immediate or alternate) shall be available from the low-voltage switching station.
 - (c) The physical design of the switchyards should minimize the probability of a single equipment failure causing the simultaneous or sequential loss of both offsite power circuits.
 - (d) The COL applicant shall connect the above transmission lines to the main transformer and reserve auxiliary transformer by separate and independent overhead lines. The COL applicant shall route these lines and their associated instrumentation and control circuits in such a way that a single event would not cause the simultaneous or sequential loss of both offsite power circuits.
 - (e) The COL applicant shall design the switchyard components to permit periodic inspection and testing in accordance with GDC 18. The design shall include the capability to periodically test the operability and functional performance of the components of the systems under conditions as close to the design function as practicable.
 - (f) Each switchyard shall have two redundant and independent 125-V dc power systems to supply 125-V dc power for all relaying, control, and monitoring equipment in the switchyards.

- (g) The breakers for each switchyard shall have redundant and independent trip circuits powered from two independent 125-V dc power sources.
- (h) Two independent protective relaying schemes shall protect each switchyard.

These eight issues were designated as DSER Open Item 8.2.1-1.

In Amendment L, ABB-CE revised CESSAR-DC Sections 8.2.1.2, 8.2.1.3, 8.2.1.6 and 8.2.2.1 to include these requirements for that portion of the offsite power systems that is outside the System 80+ scope of design certification. The staff concludes that the inclusion of the above requirements will minimize to the extent practical the likelihood of simultaneous failure of the two designated offsite sources under operating, postulated accident, and postulated environmental conditions. These provisions satisfy the independence requirements of GDC 17, and are acceptable. On this basis, DSER Open Item 8.2.1-1 is resolved.

By letter dated July 29, 1991, the staff informed ABB-CE that the grid stability criteria for the offsite power system, as discussed in Item 2 (above), did not include the loss of the largest single supply to the grid to ensure that loss does not result in the loss of grid stability and the availability of offsite power to the plant. By letter dated December 20, 1991, ABB-CE revised CESSAR-DC Section 8.2.1.6 to include this requirement. This is consistent with SRP Section 8.2.III.1.f and is acceptable.

For Item 3 (above), ABB-CE took credit for the RCP coastdown and, therefore, the COL applicant will be required to demonstrate that the effects of electrical grid disturbances on its grid are such that the limiting underfrequency decay rate of 3 Hz/sec is not exceeded. In Amendment Q, ABB-CE included this as an interface requirement.

For Item 4 (above), ABB-CE assumed a time delay of 3 seconds between a turbine trip and the LOOP for (1) feedwater line break, (2) RCP locked rotor, (3) steam generator tube rupture, (4) inadvertent opening of the steam generator atmospheric dump valve, and (5) chemical and volume control system pressure level control system malfunction with a LOOP event. Previously, the accident analysis for the System 80 designs assumed a 3-second time delay for a steam generator tube rupture and RCP locked rotor events. ABB-CE proposed a 3-second time delay between a turbine trip and a LOOP for all these events. The assumption of a time delay between a turbine trip and the LOOP was based on technical data that were derived from actual power grids.

On March 31, 1982, the applicant for CE System 80 sent the technical data on actual power grids to the NRC. ABB-CE used the information from this report to calculate the frequency response of the grid over time to a loss of a 1300-MW System 80 plant and applied the generating deficiencies caused by the loss of this plant to a model of the frequency response of the Florida electrical system to determine time to LOOP. The Florida peninsula was considered to have more power system disturbances than any other area of the United States.

In this analysis, ABB-CE modeled the loss of a System 80 plant by assuming that the grid lost 10 percent of its generating capacity when the turbine tripped. ABB-CE assumed that the plant lost offsite power instantly at a grid frequency of 57.6 Hz for all System 80 plants. The staff was concerned that the analysis may be outdated and may not represent current grid conditions and design philosophy.

The staff has always required that the time of LOOP should be assumed conservatively. Traditionally, for all accident analyses with a LOOP, the offsite power has been conservatively assumed to be lost simultaneously with the occurrence of the accident or event. In the DSER, the staff stated that ABB-CE was not being conservative in this respect. Therefore, the staff required that a 3-second time delay not be assumed between turbine trip caused by reactor trip and LOOP for all the above events, especially in the design of advanced reactors which are expected to offer enhanced safety margins for public health and safety. This was designated as DSER Open Item 8.2.1-2.

In a subsequent submittal, ABB-CE informed the staff that in order to satisfy RCP flow conditions assumed in CESSAR-DC Chapter 15, "Accident Analyses," a 3-second time delay between turbine trip and subsequent LOOP event will no longer be assumed in the System 80+ accident analysis. In Amendment Q, ABB-CE revised CESSAR-DC Section 8.2.2.1 to remove reference to a 3-second time delay between turbine trip and LOOP. Consequently, a 3-second time delay between turbine trip and LOOP is no longer assumed for accident analyses. On this basis, DSER Open Item 8.2.1-2 is resolved.

8.2.2 Offsite Power System Within the System 80+ Scope of Supply

The offsite power system for the System 80+ design includes the set of electrical circuits and associated equipment that will be used to interconnect the offsite transmission system, the main generator of the plant, and the onsite electric power distribution systems. It includes the main transformers, the unit auxiliary and reserve transformers, and the isolated phase buses, with their associated auxiliary

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systems, protection relays, and instrumentation and control.

The immediate offsite power circuit for the System 80+ design will be either (1) the main generator circuit in the backfeed direction from the transmission network through the main stepup transformer to the unit auxiliary transformers or (2) a direct feed from the reserve auxiliary transformers. The COL applicant may select the immediate offsite power source, as needed according to the reliability of switchyard sources. This aspect of the design is discussed in detail in Section 8.3.1 below. If the COL applicant selects the backfeed circuit as an immediate offsite source, this circuit will be the normal source of power for all plant loads (Class 1E, non-safety-related, and permanent non-safety-related) during all modes of operation. To initiate this backfeed, a low-voltage generator circuit breaker must disconnect from the generator. The generator circuit breaker is on the isolated phase bus between the main generator and the connection point of the unit main and unit auxiliary transformers. The generator circuit breaker will meet the guidelines of Appendix A to SRP Section 8.2. It will have the capability to interrupt the system's maximum fault current. Verification testing will include, as a minimum, all tests outlined in Section B.2 of Appendix A to SRP Section 8.2. An alternate offsite power circuit from the two dedicated reserve auxiliary transformers will be a second source of offsite power to the Class 1E loads, some non-safety-related loads, and permanent non-safety-related loads when the immediate offsite power circuit is unavailable.

If a direct feed from the reserve auxiliary transformers is an immediate offsite power source, then the backfeed circuit will be an alternate offsite power source.

Before and during startup of the nuclear unit, the power from the transmission system will be supplied through the main transformers and the unit auxiliary transformers by disconnecting the generator circuit breaker. After the unit generator attains rated speed, the unit generator will be connected to the system by closing the generator circuit breaker.

ABB-CE stated that it will maintain at least 15 meters (m) (50 feet (ft)) between the two unit auxiliary transformers and between the unit auxiliary transformer and the main transformer. The reserve auxiliary transformers will be located on the opposite side of the plant. A minimum distance of 137m (450 ft) will be maintained between the reserve auxiliary transformers and the unit auxiliary transformers.

In the DSER, the staff stated that such separation between the immediate and alternate offsite power circuit transform-

ers will minimize to the extent practicable the likelihood of simultaneous failure of both immediate and alternate offsite power circuits and was acceptable. However, since the routing of the circuits from these transformers to the onsite distribution system was not addressed, the staff needed the following additional information from ABB-CE:

- (1) the routing of the power circuits from the reserve auxiliary transformers and unit auxiliary transformers to the input terminals of the Class 1E buses, and the associated instrumentation and control circuits, to minimize to the extent practicable the likelihood of simultaneous failure of both normal and alternate offsite power circuits under operating and postulated accident and environmental conditions
- (2) the source of control power for each of the offsite power circuits that feeds the redundant Class 1E buses

These issues were designated as DSER Open Item 8.2.2-1.

Subsequently, in Amendment Q, ABB-CE revised CESSAR-DC Section 8.3.1.1.1.5, to specify routing of the power, control, and instrumentation circuits from the reserve auxiliary transformers and unit auxiliary transformers. The physical separation and electrical independence between these circuits is achieved in the following manner. The normal and alternate offsite circuits are routed from the switchyards to their respective transformers by overhead lines. The unit auxiliary transformers are separated from each other and from the reserve auxiliary transformers and from the main transformer by a minimum of 15m (50 ft). The isolated phase duct bank or cables or both located outside the turbine, control, and reactor buildings, that are affiliated with the normal offsite power circuits, are separated by a minimum of 15m (50 ft) from the reserve auxiliary transformers. Likewise, the cables located outside the turbine, control and reactor buildings that are affiliated with the alternate offsite circuit are separated by a minimum of 15m (50 ft) from the unit auxiliary and the main transformers.

Once cables enter the plant, separation of normal and alternate offsite power circuits within the nuclear annex, turbine, control and reactor buildings is maintained by fire-rated floors and walls, except within the switchgear room where they are routed on opposite sides of the room and are connected to the switchgear lineup on the opposite ends.

The instrumentation and control cables that are affiliated with the normal offsite circuits are routed in solid, metal

raceways corresponding to the load group of their power source. Similarly, the instrumentation and control cables that are affiliated with the alternate offsite circuit are routed in dedicated raceways. The alternate offsite power circuit instrumentation and control cables do not share raceways with any other cables. The separation between normal and alternate offsite power instrumentation and control cables are the same as the separation between normal and alternate offsite power cables (i.e., floors, walls, or 50 ft of physical separation) except (1) at the switchgear, where they are routed to opposite ends of the medium voltage switchgear, and (2) inside the main control room (MCR), where they are separated by routing the circuits in separate raceways.

There are no electrical interconnections between the normal and alternate offsite power, instrumentation and control circuits except where the power circuits connect to common Class 1E and non-Class 1E switchgear lineups. At the common switchgear, one open and one closed circuit breaker maintain electrical independence. These breakers are interlocked so that the closed breaker must be opened before the open breaker can be closed.

Similarly, the main generator and main generator circuit breaker power circuits are separated from the reserve auxiliary transformer power feeders by a minimum of 15m (50 ft), or by fire-rated walls or floors. Outside the MCR, the main generator and generator circuit-breaker instrumentation and control circuits are separated from the reserve auxiliary transformer's instrumentation and control circuits by a minimum of 15m (50 ft), or by walls or floors. Within the MCR, the main generator and generator circuit-breaker control and instrumentation circuits are separated from the reserve auxiliary transformer control and instrumentation circuits by routing the circuits in separate raceways.

On this basis, the provisions for separating the normal and alternate offsite power circuits and isolating the instrumentation and control cables will minimize to the extent feasible the likelihood of simultaneous failure of both normal and alternate offsite power circuits under operating, postulated accident, and postulated environmental conditions. Therefore, the independence requirements of GDC 17 are satisfied and the provisions are acceptable.

With regard to the source of control power for each of the offsite power circuit breakers that feeds the redundant Class 1E buses, ABB-CE stated that it will come from the redundant Class 1E batteries. This satisfies the independence requirements of GDC 17 and is acceptable.

On this basis, ABB-CE's approach to DSER Open Item 8.2.2-1 is acceptable and DSER Open Item 8.2.2-1 is resolved.

The applicant will design each unit auxiliary transformer to have sufficient capacity and capability to power one division of Class 1E, non-safety-related, and permanent non-safety-related loads. Each reserve auxiliary transformer will be designed to have sufficient capacity and capability to power one division of Class 1E and permanent non-Class 1E loads. In addition, each reserve auxiliary transformer will have the capacity to power one RCP and its auxiliaries.

Although the staff stated in the DSER that this design of the unit auxiliary and reserve auxiliary transformers was acceptable, ABB-CE had to submit additional information on sizing of these transformers to verify that adequate capacity will exist for the given loads. This was designated as DSER Open Item 8.2.2-2.

ABB-CE submitted additional information on sizing of unit auxiliary and reserve auxiliary transformers. The two reserve auxiliary transformers are sized to supply, within their self-cooled rating, the most conservative power requirements of its associated Class 1E buses (switchgear, load centers, and motor control centers (MCC)), the most conservative power requirements of its associated permanent non-safety bus (switchgear, load centers, and MCCs), and power requirements of at least one RCP and its support systems. Additional margins of 33-1/3 percent and 66-2/3 percent are gained by such auxiliary cooling as forced air (FA)/forced oil (FO)/forced oil and air (FOA) to allow for future load growth. Likewise, the unit auxiliary transformers are sized to supply within their self-cooled rating the most conservative requirements of its two 13.8-kV non-safety buses, one 4.16-kV bus and its associated load centers and MCCs, one 4.16-kV permanent non-safety bus and two 4.16-kV Class 1E buses with their associated load centers and MCCs. Additional margins of 33-1/3 percent and 56-2/3 percent are gained by such auxiliary cooling as FA/FO/FOA to allow for future load growth.

In Amendment Q, ABB-CE revised CESSAR-DC Sections 8.1.3.B.5 and 6 to include the additional information. On this basis, the staff concludes that the unit auxiliary and reserve auxiliary transformers will have sufficient capacity and capability to ensure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded as a result of anticipated operational occurrences and (2) the core will be cooled, and containment integrity and other vital functions will be maintained in the event of postulated accidents. Therefore, the normal and alternate offsite power trans-

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formers will satisfy the capacity and capability requirements of GDC 17 and are acceptable. On this basis, DSER Open Item 8.2.2-2 is resolved.

In CESSAR-DC Section 8.2.1.4, ABB-CE states that all systems, equipment, and components associated with the immediate and alternate offsite power circuits have the capability of being tested during plant operation. However, in the DSER, the staff asked ABB-CE to include the following information to ensure that the requirements of GDC 18 with regard to testing and inspection of electric power systems important to safety are satisfied:

- (1) List the components of the offsite power system that can be tested during power operation and describe the rationale for components that are not tested during power operation.
- (2) Describe the process for testing the functional capability and for calibrating the instrumentation, control, and protection systems associated with the normal and alternate circuits of the offsite power system.
- (3) Describe the process and frequency of periodically verifying that the alternate offsite power circuit is energized and is connected to the appropriate Class 1E distribution to ensure its availability.

This was designated as DSER Open Item 8.2.2-3.

In Amendment Q, the ABB-CE revised CESSAR-DC Section 8.2.1.4 to include systems, equipment, and components associated with the offsite power circuits that will not be tested during power operation and gave the rationale for components that are not tested during power operation. All systems, equipment, and components associated with the immediate and alternate offsite power circuits, testing of the generator breaker and transfer of power from normal to alternate offsite power source will be tested during startup and shutdown and not during power operation. The instrumentation, control, and protection systems associated with the normal and alternate offsite circuits will be periodically calibrated to perform their required function during power operation. The switchyard systems and components will be periodically tested and inspected during shutdown.

Detailed test procedures will be developed for testing the functional capability and for calibrating the instrumentation, control, and protection systems of the immediate and alternate circuits of the offsite power systems. The design of high- and medium-voltage bus ducts gives ready access for regularly inspecting, cleaning, and tightening terminals, and for regularly inspecting and cleaning insulators.

Breaker alignment and indicated power availability for each of the required offsite power circuits will be verified once every 7 days to ensure that the normal and alternate offsite power circuits are energized and connected to the appropriate Class 1E bus.

The test and inspection intervals for systems, equipment, and components of the offsite power systems will be established and maintained in accordance with the guidelines of IEEE 338-1977, Section 6.5.

On this basis, the testability requirements for systems, equipment, and components associated with immediate and alternate offsite power systems in the System 80+ design will ensure that electrical systems and components important to safety are designed to permit appropriate periodic inspection and testing. This satisfies the periodic testing and inspection requirements of GDC 18 and is acceptable. On this basis, DSER Open Item 8.2.2-3 is resolved.

8.2.3 Grounding

The CESSAR-DC did not address the system design and performance requirements for the plant grounding systems and surge protection systems. By letter dated July 29, 1991, the staff requested a discussion of lightning protection for the main, unit auxiliary, and reserve auxiliary transformers.

By letter dated June 14, 1992, ABB-CE stated that the COL applicant will protect the main, unit auxiliary, and reserve auxiliary transformers from lightning. ABB-CE also stated that the lightning protection for dry-type transformers will be considered in the final design.

In the DSER, the staff stated that this response was inadequate and required ABB-CE to address the following Electric Power Research Institute (EPRI) guidelines regarding plant grounding and lightning protection:

- (1) The plant grounding grid, consisting of bare copper cables, shall limit the step and touch potentials to safe values under all fault conditions.
- (2) The grounding system shall have bare copper risers for all electrical underground ducts and equipment, and for connections to the grounding systems within buildings.
- (3) The design and analysis of the grounding system shall follow the procedures and recommendations of the latest revision of IEEE 665.
- (4) Each building shall have grounding systems connected to the plant grounding grid. As a minimum,

every other steel column of the building perimeter will be connected directly to the grounding grid.

- (5) The main generator shall be grounded with a neutral grounding device having an impedance that shall limit the maximum phase current under short circuit conditions to a value not greater than that for a three-phase fault at its terminals.
- (6) Suitable grounding device(s) shall ensure proper grounding of the isophase buses when the generator is disconnected from the isophase buses.
- (7) The onsite medium voltage ac distribution system shall be resistance grounded at the neutral point of the low-voltage windings of the unit auxiliary and reserve transformers.
- (8) The neutral point of the generator windings of the onsite standby power supply units, Class 1E and non-Class 1E, shall be grounded through distribution-type transformers and loading resistors sized for continuous operation with a ground fault.
- (9) The neutral point of the low-voltage ac distribution systems shall be either solidly or impedance grounded, as necessary to properly coordinate ground fault protection.
- (10) The dc systems shall be ungrounded.
- (11) Each major piece of equipment, metal structure, or metallic tank shall have two ground connections diagonally opposite each other.
- (12) The ground bus of all switchgear assemblies, MCCs, and control cabinets shall be connected to the plant ground grid through at least two parallel paths.
- (13) One bare copper cable shall be installed with each underground electrical duct run, and all metallic hardware in each personnel hatch shall be connected to this cable.
- (14) Plant instrumentation shall be grounded through separate radial grounding systems, consisting of isolated instrumentation ground buses and insulated cables. The instrumentation grounding systems shall be connected to the plant grounding grid at one point only and shall be insulated from all other grounding circuits.

- (15) The plant analog and digital instrumentation systems shall have separate instrumentation grounding systems.
- (16) All major plant structures, including the containment enclosure building, shall be protected from lightning. The lightning protection shall be in accordance with the National Fire Protection Association's Lightning Protection Code, NFPA-78, and the Nuclear Energy Property Insurance Association's Basic Fire Protection for Nuclear Power Plants.
- (17) Each phase of all tie lines connecting the plant electrical systems to the switching station(s) and offsite transmission system shall be protected by lightning arresters. These arresters will be connected to the high voltage terminals of the main stepup and reserve transformers.
- (18) Plant instrumentation and monitoring equipment located outdoors or connected to outdoor cables shall have built-in surge-suppression devices to protect the equipment from lightning-induced surges.

These issues were designated as DSER Open Item 8.2.3-1.

In Amendment Q, ABB-CE added Section 8.3.1.1.9 to the CESSAR-DC, "Grounding and Lightning Protection Criteria." This new section addresses compliance with all the EPRI guidelines regarding plant grounding and lightning protection with the exception of Item 15 (above). For this item, ABB-CE stated that separate ground buses for digital and analog instrumentation system are not necessary or practical because many vendor-supplied systems contain both analog and digital components in the same cabinets and they do not supply separate ground buses for these components. Further, separate buses would require a significant amount of cabling and more congested cabinets. Also, IEEE 1050-1989, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations," does not require separate ground buses for analog and digital systems. This justification is acceptable. Further, the inclusion of each of the EPRI plant grounding guidelines with the exception of Item 15 will ensure that plant structures, systems, and equipment will be appropriately grounded and protected from lightning. Therefore, this design is consistent with the protection and independence requirements of GDC 17 and is acceptable. On this basis, DSER Open Item 8.2.3-1 is resolved.

8.3 Onsite Power Systems

8.3.1 Onsite Class 1E AC Power System

The onsite Class 1E ac power system includes those standby power sources, distribution systems, and vital supporting systems that supply power to Class 1E equipment. The safety function of the ac onsite power system is to provide capacity and capability sufficient to ensure that the structures, systems, and components important to safety perform as intended. The staff's objective is to determine whether the ac onsite power system has the required redundancy, meets the single-failure criterion, is testable, and has the capacity, capability, and reliability to supply power to all required safety loads, in accordance with GDC 2, 4, 17, and 18.

The onsite Class 1E ac power system consists of two redundant and independent 4.16-kV engineered safety feature (ESF) distribution systems, each with their 480-V load centers and MCCs, 120-V ac vital power system, and the emergency ac power supplies. Each of the two 4.16-kV safety buses consists of two buses (A and C for Division I and B and D for Division II). In the original design, the immediate offsite power source for the 4.16-kV ESF buses was back-fed through the main stepup transformer to the unit auxiliary transformers by opening the generator breaker to disconnect the generator from the transmission network. The alternate offsite power source to the 4.16-kV ESF buses was through two reserve auxiliary transformers, each having a dual winding secondary, one winding of which fed one 4.16-kV safety bus through its associated permanent non-Class 1E bus, and the other winding fed two of the 13.8-kV non-safety buses. The reserve auxiliary transformer primary winding was connected directly to the low-power switchyard (preferred switchyard interface II).

By letter dated July 29, 1991, the staff informed ABB-CE that feeding the safety buses from the offsite power sources through non-safety buses was not the most reliable configuration. In this configuration, the safety loads are subjected to transients caused by the non-Class 1E loads and add additional failure points between the offsite power sources and safety loads. To overcome these shortcomings, the staff recommended energizing the safety buses directly from the two reserve auxiliary transformers. The staff also recommended retaining the supply from the unit auxiliary transformer through the permanent non-safety buses to the safety buses as a manually switched alternate power supply. Powering safety buses directly from offsite power source is an NRC policy issue (SECY-91-078, Section II.B of Enclosure 1 to SECY-93-087).

In SECY-91-078, the staff recommended that the Commission approve the staff position that an evolutionary plant design should include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus. In its August 15, 1991, SRM, the Commission approved the staff's position.

By letter dated December 20, 1991, ABB-CE revised its design to add a direct feed from the reserve auxiliary transformers to each of the Class 1E buses. ABB-CE retained a manual feed from the permanent non-safety buses to the safety buses. This satisfies the requirements of this policy issue and the staff's proposed applicable regulation and is acceptable.

ABB-CE later advised the staff that it would allow the COL applicant to choose whether to normally feed the safety buses from the reserve auxiliary transformers or through permanent non-safety buses. The COL applicant would make the choice after considering the specific needs at the site, including the voltage and reliability of the switchyard source. If the circuits from the reserve auxiliary transformers are not selected as the normal offsite power source to the redundant Class 1E buses, these connections to the redundant Class 1E buses will be alternate offsite power feeds. The COL applicant would manually connect a safety bus to its alternate power supply circuit.

Although the staff preferred to have safety buses normally fed directly from the offsite source, the staff also recognized that this design feature must be viewed in the context of the overall design of the electrical system design, and that some of the design concepts and objectives, such as the three-tier concept and the objective to simplify the design, affect the choices. In the DSER, the staff concluded that either operating mode satisfied the requirements of GDC 17, regarding two physically independent circuits from the transmission network to the onsite distribution system and, therefore, either could be acceptable. However, the COL applicant needed to specify normal and alternate modes selected when it specifies the design requirements for offsite power systems. This was designated as COL Action Item 8.3.1-1.

Subsequently, in Amendment Q, ABB-CE added a new section, CESSAR-DC Section 8.1.4.5, "COL Action Items," to address COL action items. Item A of this section requires the COL applicant to specify the immediate and alternate sources of offsite power for the 4.16-kV Class 1E buses based on its site reliability analysis of

normal and alternate offsite power sources. This satisfies the staff's requirement and is acceptable.

The Class 1E onsite power systems have sufficient capacity and capability to mitigate the effects of an accident assuming LOOP. The two divisions of the Class 1E onsite power and distribution systems are independent. In accordance with the guidance of RG 1.6, the loads and sources will not be transferred between redundant safety buses.

The redundant Class 1E onsite power systems in the System 80+ design will be located in seismic Category I structures. The Class 1E switchgear, load centers, MCCs, and the emergency diesel generators (EDGs) of redundant load groups will be located in separate rooms of seismic Category I buildings that protect against the effects of natural phenomena and missiles. This condition satisfies the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," which requires that Class 1E systems, equipment, and components be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. The Class 1E systems have been designated to be seismically and environmentally qualified to withstand the environmental conditions associated with normal operation and postulated accidents. This satisfies the requirement of GDC 4, "Environmental and Dynamic Effects of Design Bases," which requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents.

In CESSAR-DC Section 8.3.1.1.7, ABB-CE requires that all the Class 1E 4.16-kV safety division breakers be equipped with redundant trip coils and with two sources of control power, one for each redundant circuit. By letter dated July 29, 1991, the staff requested that ABB-CE list the sources of control power supply for these breakers, including the separation criteria.

By letter dated January 14, 1992, ABB-CE responded that having redundant trip coils for Class 1E 4.16-kV safety-related circuit breakers increases the reliability of the 4.16-kV breaker trip function. One of the power sources would be from the Division I battery and the other would be from the associated channel battery of the same division. ABB-CE stated that since these two batteries are in the same division separation between the two trip coil circuits is not required. In the DSER, the staff stated that one coil could be powered from a channel of a different color and, in that case, separation would be required. Therefore, the staff required that ABB-CE clarify this design to the staff. This was designated as DSER Open Item 8.3.1-1.

In Amendment Q, ABB-CE clarified the power sources for the redundant trip coils for Class 1E 4.16-kV safety-related circuit breakers. The power source for the Division I Class 1E 4.16-kV circuit-breaker redundant trip coils is from the Division I battery and the Channel A battery, which do not require separation (both are red). Likewise, the power sources for the Division II Class 1E circuit breaker redundant trip coils are from the Division II battery and the Channel B battery, which do not require separation (both are green). Since the control power supply of the redundant trip coils for Class 1E 4.16-kV circuit breakers are from the same division batteries, no separation would be required. This resolves the staff's concern and is acceptable. On this basis, DSER Open Item 8.3.1-1 is resolved.

8.3.1.1 Non-Class 1E AC Power Systems

The onsite power distribution system for the System 80+ design is arranged in three tiers. The first tier of systems consists of the distribution systems feeding non-safety-related loads required exclusively for unit operation. These systems can be fed from the main generator and from the offsite power system, through a backfeed configuration, when the main generator is unavailable. These systems can also be fed through the reserve auxiliary transformer to power a large load on these buses.

The second tier consists of the distribution systems supplying power to permanent non-safety-related loads that are generally required to remain operational at all times. These loads are normally fed from the same source that feeds the first-tier loads, but they can also be fed from a second independent offsite source or a combustion turbine generator (CTG) if their normal power source is unavailable.

The third tier consists of the distribution systems feeding the Class 1E loads as discussed in Section 8.3.1 herein.

The non-Class 1E ac power system consists of the unit main turbine generator, associated isolated phase bus, the generator circuit breaker, four single-phase unit main transformers (one spare), two 50-percent-capacity unit auxiliary transformers, and one 50-percent-capacity unit auxiliary transformer installed as a spare. Each unit auxiliary transformer has enough capacity to power one set of non-safety-related loads, permanent non-safety-related loads, and Class 1E loads.

The 13.8-kV normal auxiliary power system (first tier) consists of four (x1, x2, y1, and y2) non-safety switchgears. Two auxiliary transformers, each with dual low-voltage 13.8-kV/4.16-kV secondary windings, supply the four 13.8-kV switchgears. One low-voltage winding

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of each transformer in this arrangement normally supplies two 13.8-kV switchgears (x1 and x2 from unit auxiliary transformer "x," and y1 and y2 from unit auxiliary transformer "y"), and the other winding feeds 4.16-kV non-safety and permanent non-safety buses. The 13.8-kV switchgear supplies power to such large motors as the RCP motors and condensate water pump motors. If the switchgears lose their normal power, selected loads (at least two RCPs) on the 13.8-kV buses can be fed manually from the reserve auxiliary transformers (one RCP from each reserve auxiliary transformer), which are connected to preferred switchyard interface II. The design permits manual start and operation of a large load connected to these buses, such as an RCP or a condensate pump to bus x1 or x2 and bus y1 or y2, if the normal supply is lost or unavailable. Alternate source of power for non-safety-related loads is an NRC policy issue (SECY-91-078, Section II.B of Enclosure 1 to SECY-93-087).

In SECY-91-078, the staff recommended that the Commission approve the staff's position that an evolutionary plant design should include an alternate power source to the non-safety-related loads, unless it can be demonstrated that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs. In its August 15, 1991 SRM, the Commission approved the staff's position. The staff, in its safety evaluation report (SER) for the EPRI Evolutionary Utility Requirements Document (URD) clarified the intent of this position: "...an alternate power source be provided to a sufficient string of non-safety loads so that forced circulation could be maintained, and the operator would have available to him the complement of non-safety equipment that would most facilitate his ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip."

The System 80+ design satisfies the NRC policy as stated in SECY-91-078 and Section II.B of Enclosure 1 to SECY-93-087 and the staff's proposed applicable regulation and is, therefore, acceptable.

The 4.16-kV normal auxiliary power system (second tier) consists of four switchgear groups. The first switchgear group, designated non-safety "x," is connected to unit auxiliary transformer "x" to power large loads such as the turbine building cooling water system pumps. The second switchgear, designated non-safety "y," is connected to unit auxiliary transformer "y" to power similar non-safety-related loads.

The third switchgear group, designated permanent non-safety "x," powers loads that must remain operational at all times because of their specific functions. Its normal

source of power is unit auxiliary transformer "x." If this switchgear loses its normal source, it will be automatically transferred to the reserve auxiliary transformer. If the power is lost from both the unit auxiliary transformer and the reserve auxiliary transformer, selected permanent non-safety-related loads will be automatically transferred to a CTG, a standby non-safety power source. This CTG is also used as the alternate ac (AAC) source for station blackout (SBO).

The fourth switchgear group, also designated permanent non-safety "y," is normally connected to unit auxiliary transformer "y." If the normal source for this switchgear is lost, this switchgear will also automatically transfer to the reserve transformer. If power is lost from both the unit auxiliary transformer and the reserve transformer, selected permanent non-safety-related loads will be automatically transferred to the CTG.

8.3.1.2 Alternate AC Source

The AAC is a non-safety-related CTG provided to cope with a LOOP or an SBO scenario. The AAC source is of sufficient size to accommodate either of the following loading conditions:

- (1) both sets of "x" and "y" permanent non-safety-related loads or
- (2) one set of permanent non-safety-related loads and one set of a safety division's loads as indicated below:
 - (a) permanent non-safety "x" and Division I only or
 - (b) permanent non-safety "y" and Division II only

Although these requirements specify the divisions that the AAC source is required to supply, ABB-CE did not specify which scenarios apply to these requirements and, therefore, which complement of loads it must supply within those divisions. By letter dated July 29, 1991, the staff recommended that, as a minimum, the AAC source be specified to power, the worst-case shutdown (to cold shutdown) or accident loads (whichever is greater) within the complement of divisions specified above. The staff also recommended that the AAC source should have the capability to power those loads with some margin for load growth, when operating within its continuous rating.

By letter dated January 14, 1992, ABB-CE agreed to require that the AAC source be capable of powering the worst-case shutdown or design-basis accident loads to cold shutdown conditions when operating within its continuous rating, as recommended by the staff. ABB-CE will

compensate for the growth of the load by adding a margin of 10 percent to the continuous rating of the AAC source.

The AAC source is designed to start automatically within two minutes from the beginning of a LOOP event. The CTG is available for loading if either of the 4.16-kV permanent non-safety buses becomes de-energized. A sequencer similar to that used for Class 1E buses automatically sequences the loading of the "x" or "y" permanent non-safety-related loads or of both. Sufficient physical and electrical separation is maintained between the AAC source circuits and the offsite power circuits so that a failure in one system will not affect the operation of the other circuit. A non-safety battery powers the instrumentation and controls necessary to start and run the CTG. In the DSER, the staff concluded that this design conformed with RG 1.155 which recommends that the AAC source should be available in less than 1 hour after the onset of SBO and that there should be a minimum potential for common cause failure with the offsite or the Class 1E EDGs and was acceptable. However, ABB-CE needed to address how the power, control, and instrumentation circuits associated with the AAC source were routed to ensure adequate physical and electrical separation exists between Class 1E circuits and circuits associated with the AAC source. This was designated as DSER Open Item 8.3.1.2-1.

In Amendment Q, ABB-CE submitted the needed information. The AAC source is located in a separate area inside the plant protected area in a self-contained enclosure. The "x" and "y" circuits are routed in the turbine building in a dedicated raceway (conduit/ductline/cable tray). Physical separation between these circuits is provided so that a single failure will not affect both circuits. Once in the turbine building, the AAC circuits are routed directly to the two permanent non-safety switchgears ("x" and "y"). The permanent non-safety switchgears are located in the turbine building which contains only non-Class 1E circuits. There are no Class 1E cables in this area. The CTG and the offsite power circuits are separated as follows: The circuits associated with the CTG and offsite power circuits are routed on different floors within the turbine building to the permanent non-safety buses with the "x" and "y" circuits separated by routing in different areas or on opposite walls to maintain separation. These circuits enter common switchgear from the opposite ends, thus maintaining separation. There are no electrical interconnections between the normal offsite and alternate offsite, power, control, and instrumentation circuits except where the power circuits connect to common Class 1E and non-Class 1E switchgear lineups. At the common switchgear, one closed and one open circuit breaker maintain electrical independence. The circuit breakers are interlocked so that the closed breaker must be opened before the open breaker

can be closed. Similarly, the power, instrumentation, and control circuits of the AAC source are electrically independent from that of the normal and alternate offsite power circuits.

These provisions for separation and electrical independence of circuits associated with normal and alternate offsite power systems and the CTG minimize to the extent feasible the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. This satisfies the recommendations of RG 1.155 with regard to having a minimum potential for common cause failure of the offsite and the AAC sources and is acceptable. On the basis, DSER Open Item 8.3.1.2-1 is resolved.

8.3.1.3 Use of the AAC Source To Satisfy Technical Specification Requirements

In CESSAR-DC Section 8.3.1.1.2.1, ABB-CE states that, if an EDG is out of service or has failed, the AAC source can be aligned to supply emergency power to either Class 1E bus. By letter dated July 29, 1991, the staff informed ABB-CE that although the AAC power source may have the capability to supply one division of safety loads, the staff has not yet determined to what extent, if any, the AAC source could be used to satisfy technical specification (TS) requirements to supply emergency power to either Class 1E bus if a EDG is out of service or has failed.

By letter dated December 20, 1991, ABB-CE responded that, although it states in CESSAR-DC Section 8.3.1.1.2.1 that the CTG is provided to help cope with the effects of LOOP and SBO scenarios, the AAC source is not included in the proposed TS in CESSAR-DC Chapter 16. It should be noted that the staff will permit the use of the CTG as a surrogate for a EDG under reduced inventory conditions. This has been evaluated as part of the shutdown risk evaluation as documented in Chapter 19 of this report and through the evaluation of the System 80+ technical specifications.

Subsequently, ABB-CE proposed that the AAC source be used to satisfy TS requirements when a EDG is out of service at power. The staff has evaluated this aspect of the design as part of its review of the TS described in Chapter 16 of this report.

8.3.1.4 Emergency Onsite Power Sources (Diesel Generators)

Two EDG units supply onsite Class 1E emergency power. Both units are automatically started by either a safety injection actuation signal (SIAS) or a 4.16-kV Class 1E

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bus undervoltage signal. Each EDG is connected to its respective safety buses (A and C for Division I and B and D for Division II) through two normally open breakers. There is one EDG for each division. Each EDG with its auxiliaries is located in a separate seismic Category I room. The EDGs have sufficient capacity to operate the safety loads needed to maintain the plant in a safe condition if a loss-of-coolant accident (LOCA) occurs concurrently with a LOOP. ABB-CE also provided a capacity margin of 10 percent in calculating the size of the EDGs to accommodate the load growth expected to occur over the life of the plant.

However, in determining the size of the EDG, ABB-CE did not apply conservative estimates of the load power factors of individual loads. This was designated as DSER Open Item 8.3.1.4-1.

In Position C.1.2 of RG 1.9, the staff recommends that, if the designer does not have fully substantiated performance characteristics for such mechanical equipment as pumps, it should calculate the electric motor drive ratings using conservative estimates of these characteristics, such as pump runout conditions and motor efficiencies of 90 percent or less and power factors of 85 percent or less. In determining the rating of the EDG for the System 80+ design, ABB-CE used a power factor of 90 percent, instead of 85 percent as recommended by the guide.

In the DSER, the staff required that ABB-CE reevaluate the size of the EDGs using a power factor of 85 percent for individual loads, as recommended by the guide.

Subsequently, ABB-CE agreed to use a power factor of 0.85 rather than 0.90 for calculating the motor drive ratings for such mechanical equipment as pumps, which gives higher equivalent kilovolt-ampere (kva) in sizing the EDGs. Likewise, in order to size the engine, the loads listed in kva will be converted to an equivalent Kilowatt (kW) by multiplying by a power factor of 0.90. This also gives a higher equivalent kW and, therefore, a more conservative Kw for engine sizing. On this basis, the size of the diesel engines and generators for the System 80+ design is consistent with RG 1.9, Position C.1.2 and is acceptable. On this basis, DSER Open Item 8.3.1.4-1 is resolved.

Each EDG is designed to attain the rated voltage and frequency within 20 seconds and to be ready to sequence loads after receiving a start signal to meet the response times assumed in the accident analysis. During the loading sequence the frequency and voltage are maintained at 95 percent and 75 percent of nominal, respectively. This is consistent with RG 1.9, Position C.1.4 which recommends that the diesel generator design should be such that no time

during the loading sequence should the frequency decrease to less than 95 percent of nominal nor the voltage decrease to less than 75 percent of nominal, and is acceptable.

In the DSER, the staff required ABB-CE to submit the dynamic analysis of loading time intervals between various load groups for the staff to review. This was designated as DSER Open Item 8.3.1.4-2.

In CESSAR-DC Table 8.3.1-2, ABB-CE states the EDG sequencing loading time interval between various load groups. By letter dated July 29, 1991, the staff expressed concern that the loading time interval of 2.5 seconds is not sufficient for the first three groups of loads. In general, large induction motors can achieve rated speed in less than 5 seconds when powered by EDG units of adequate size. However, these motors may not be able to achieve rated speed in 2.5 seconds. Therefore, the staff was concerned that any setpoint drift in the sequencer which reduces this short interval could cause overlapping of the other group loads and could prevent the EDG from recovering frequency and voltage within the guidelines of RG 1.9.

By letter dated January 14, 1991, ABB-CE stated that it performed a EDG dynamic analysis using data for an existing EDG and loads equivalent to those listed in the System 80+ design. ABB-CE verified that a similar EDG will function properly under the assigned loads intervals and that the frequency never dropped below 95 percent and the generator voltage never dropped below 75 percent of the nominal.

Subsequently, ABB-CE described a dynamic analysis for an existing nuclear station's diesel generators. The load sequencer for the System 80+ design utilizes solid-state relays which are not subject to drift characteristics of older load sequencer types. Regarding the relatively short interval of 2.5 seconds between the starting of large motors, ABB-CE informed the staff that the EDG load sequencers at existing plants have similar clusters of large motor loads at the beginning of their load sequence with 1-second separation. These EDGs have successfully performed within the parameter requirements of RG 1.9, both for dynamic analysis and in actual test situations. Therefore, the 2.5-second loading time interval between load groups will not cause overlapping of the affected load groups which could prevent the EDG from recovering frequency and voltage within the guidelines of RG 1.9. Having reviewed the simulated transient analysis of the EDG loading for Division I and Division II of the System 80+ design, the staff concludes that the information submitted by ABB-CE demonstrates that the voltage and frequency can be recovered within the guidelines of RG 1.9. On this basis, DSER Open Item 8.3.1.4-2 is resolved.

The staff has taken the position that (1) the EDG protective trips shall be bypassed when the EDG is required for a design-basis event if the operator has sufficient time to react appropriately to an abnormal condition of the EDG unit and (2) the protective trips shall be in force during EDG periodic testing. EDG overspeed and generator differential trips are the only exceptions allowed to this position. In retaining any other trips, ABB-CE must use coincident logic to avoid spurious trips. The System 80+ design includes these two trips and the low-low lube oil pressure and generator voltage-controlled overcurrent, using two or more independent measurements with coincident logic under emergency operation. The design of the bypass circuitry is consistent with IEEE 603-1980 and RG 1.9. All protective trips that would normally trip the EDG are alarmed in the MCR, even if they are bypassed during an accident condition. The COL applicant will provide for testing the status and operability of the bypass circuits. This is designated as COL Action Item 8.1.4.5.H. The trip bypass function can only be reset manually. This is consistent with RG 1.9, Position C.1.8, which recommends that the diesel generator protective relay bypass system design should include the capability for testing the status and operability of the bypass circuitry, for alarming in the control room for abnormal values of all bypass parameters, and for resetting the trip bypass function manually and is acceptable.

In the original CESSAR-DC, the staff interpreted that each safety division has two load sequencers. ABB-CE indicates that one load sequencer is used when an engineered safety features actuation system (ESFAS) is generated and offsite power is the source of power to an emergency bus. A separate load sequencer is used when an ESFAS is generated and a EDG is powering the emergency buses configured in a redundant pair. The load sequencer for the EDG continuously monitors the Class 1E 4.16-kV Division I and Division II buses for an undervoltage condition. If these buses receive insufficient voltage, the safety logic will start the EDG, begin shedding loads, and begin powering the loop loads. The load sequencer for the EDG assumes that a LOCA occurs prior to, concurrent with, or any time after the LOOP. In the DSER, the staff concluded that this design meets the intent of RG 1.9 and was acceptable. However, the staff needed additional information on the following topics to complete its review in this area: (1) the rationale for using a load sequencer when offsite power was available, (2) the separation between the two load sequencers, and (3) their mode of operation. This was designated as DSER Open Item 8.3.1.4-3.

In Amendment Q, ABB-CE made it clear to the staff that each safety division has only one load sequencer. Thus, there are not separate sequencers for sequencing loads when offsite power is available and when onsite power is

available. A load sequencer is used when offsite power is available, to prevent a large voltage dip when all required Class 1E loads are started by an accident signal. Without the load sequencer, oversized transformers are required to lessen the voltage dip; this is not good engineering practice. However, when the offsite power is available, the load sequence used is an accelerated sequence, which allows the voltage to return to 95 percent before sequencing the next load group. The key difference between the two sequences is that when normal offsite power is available there is no load shed.

The sequencer is a part of the ESF component control system (CCS) and is located in the same control cabinet as the ESF-CCS. The ESF-CCS controls all in-plant power paths for both the onsite and offsite sources. Each ESF-CCS has redundant internal processors and also redundant data highways that connect to the individual processors at the loads. One of the ESF-CCS's two internal processors runs the sequencing program while the other remains in a standby condition. If one processor fails, the other takes over control of the load sequencing from the point where the first left off. ABB-CE stated that there are no credible electrical faults that can leave both processors for a single division's sequencer unable to perform their function. Each of the two division ESF-CCS load sequencers is located in a physically separated Category I structure, thus satisfying the independence requirements of GDC 17 which requires that the Class 1E systems have sufficient independence and redundancy to perform their safety functions assuming a single failure.

It is a typical practice to sequence loads when offsite power is available and to use a single sequencer for both offsite and onsite power systems. If a single sequencer is used for sequencing loads when offsite or onsite power is available, the staff has required assurance that there is no credible common-mode failures in the sequencer design that could render both onsite and offsite power sources unavailable. The System 80+ design satisfies this staff requirement by providing redundant internal processors that run the sequencing program. This design is acceptable. On this basis, DSER Open Item 8.3.1.4-3 is resolved.

By letter dated July 29, 1991, the staff expressed concern that during surveillance testing when EDGs are manually synchronized to the offsite power source, an operator could accidentally attempt to close the EDG breaker either too late or too early, which could damage the EDG because the EDG would be out of phase with the offsite power source. The staff stated that including a synchronizing check relay would prevent the operator from closing the EDG breaker when the two sources are out of phase.

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By letter dated January 14, 1992, ABB-CE responded that it will include a synchronizing check relay which will limit the operator in synchronizing the EDG during surveillance testing. This relay allows closing the EDG breaker only when the phase angle difference between the EDG and offsite power system is within an acceptable range and the voltage difference between the EDG and offsite power system is within acceptable limits. This design satisfies the staff's concern and is acceptable.

Normally, during periodic testing of a EDG, the EDG is paralleled with the offsite power system. If offsite power is lost during such a test, the EDG would probably attempt to power the bus and the offsite power system through the closed offsite power feeder breaker. In this case, the EDG breaker will trip on overcurrent and, in some designs, the diesel generator breaker will lock out for this condition. By letter dated July 29, 1991, the staff required that the EDG breaker not be locked out for such overload conditions. This will ensure that the EDG will continue to be available.

By letter dated January 14, 1992, ABB-CE committed to delete locking out of the EDG breaker on initiation of generator voltage-controlled overcurrent relay from the list of trips which lock out the EDG breaker to comply with the staff's position. In the DSER, the staff concluded that ABB-CE must revise the CESSAR-DC to delete this design feature. This was designated as Confirmatory Item 8.3.1.4-1.

In Amendment Q, ABB-CE made it clear to the staff that it will retain the protective trip described above, i.e., the EDG breaker will be locked out on initiation of generator voltage-controlled overcurrent relay (51VC) during an accident. However, in order to satisfy the staff's concern, ABB-CE stated that during periodic testing, the EDG will be available for a LOCA, even if a 51VC relay is actuated. The 51VC relays are provided for protection against faults external to the generator (i.e., the 4,160-V bus). These relays are coordinated with the 4,160-V feeder breakers so that the feeder breakers will clear downstream faults before the relay actuates. Keeping the overcurrent relays in service during a LOCA allows the generator to be disconnected from an external fault, preventing damage to the EDG, and keeping the EDG available for later use by the operator.

The EDG circuit breakers are designed to trip during an accident as well as during periodic testing upon activation of the overcurrent relays (51VC). During an accident condition, the initiation of two or more independent measurements with coincident logic relay activates a lockout relay which trips the EDG breaker. This lockout relay must be manually re-set. However, during testing,

any one relay will trip the EDG breaker via a test lockout relay. This lockout relay is electrically resettable by an ESF signal so that the EDG continues to be available if it is needed for an accident condition. This satisfies the staff's concern and is acceptable. On this basis, Confirmatory Item 8.3.1.4-1 is resolved.

The design includes local and MCR alarms for each EDG. ABB-CE stated that it will determine the number of MCR alarms and the words to be placed on the alarm windows for each EDG annunciator after the NRC certifies the System 80+ design. This aspect of the design will be verified as part of the verification and validation effort of the MCR design.

In the CESSAR-DC, ABB-CE also describes conditions that can render the EDG units incapable of responding to an automatic emergency start signal. These alarms are not activated by any other alarm signals. The staff reviewed this information and concludes that each condition that can render a EDG unit incapable of responding to an automatic emergency start is alarmed in the MCR. This aspect of the design conforms with RG 1.47 with regard to providing bypass and inoperable status indication for engineered safety feature systems and is acceptable.

8.3.1.5 Testing

In the System 80+ design, the COL applicant will test and inspect Class 1E as well as non-Class 1E electrical distribution systems and components in accordance with a periodic testing program. The COL applicant will periodically test the generator power circuit breaker. The 13.8-kV and 4.16-kV circuit breakers and associated equipment can be tested in service where testing does not interfere with the operation of the unit. These breakers can be "racked out" to a test position and tested without energizing the circuits.

The testing of the 480-V circuit breakers includes injecting the primary current, individually checking each component (such as charging motors, shunt trip coils, draw out devices, heaters, air filters, wiring and connectors), and testing the resistance of contacts.

The COL applicant will also routinely test transfers to the various power sources to verify that these systems can operate properly.

The COL applicant will periodically test each protective relay under simulated conditions to verify correct operation. The COL applicant will test relay sensors such as the current transformers and potential transformers before installing them and will test them periodically thereafter.

The COL applicant will periodically test the EDGs in conformance with RG 1.108 and NRC Generic Letter 84-15, "Proposed Staff Actions To Improve and Maintain Diesel Generator Reliability."

In the DSER, the staff required ABB-CE to specify COL action items for all periodic testing commitments that were not covered by technical specifications. This was designated as Confirmatory Item 8.3.1.5-1.

In Amendment Q, ABB-CE added a new CESSAR-DC Section 8.1.4.5 on COL action items. Item B of this section requires a COL applicant to specify all items with periodic testing commitments not covered by the technical specification. This satisfies the staff's requirement and is acceptable. On this basis, Confirmatory Item 8.3.1.5-1 is resolved.

8.3.1.6 Identification of Class 1E Electrical Systems

In the CESSAR-DC, ABB-CE lists the criteria for identifying electrical equipment needed to preserve the independence of redundant equipment. All cables, raceways, and equipment are identified according to the particular safety division train or channel with which they are associated. Each Class 1E cable and raceway is identified by a color code according to its separation group. Each piece of equipment will have affixed a nameplate of an appropriate color background.

Each Class 1E cable is marked with the appropriate color code at intervals not to exceed 1.5m (5 ft) and is also identified by tags affixed at both ends and bearing the appropriate cable number. Class 1E cable trays and major pieces of equipment also have color-coded tags.

All cable trays, conduits, and wireways containing Class 1E cables are color coded for ease of identification. These raceways are marked at each end, at all entrances and exits to rooms, and at intervals not to exceed 4.5m (15 ft).

By letter dated July 29, 1991, the staff requested information on the marking of cables inside of cabinets and panels.

By letter dated January 14, 1992, ABB-CE responded that the COL applicant need not place color codes on component wires of a cable inside a cabinet or panel because each safety-related cabinet and panel will contain components for only one channel. This would imply that no cabinets or panels in the System 80+ design contain redundant cables. In the DSER, the staff concluded that ABB-CE should clarify this aspect of the design or list the criteria for identifying redundant Class 1E cables inside cabinets and panels. ABB-CE should also list the criteria for identifying associated circuits in the System 80+

design. This was designated as DSER Open Item 8.3.1.6-1.

In Amendment L, ABB-CE revised CESSAR-DC Section 8.3.1.3 to make it clear to the staff that each safety-related cabinet and/or panel contains circuits and components of only one channel. In control panels where cables from more than one channel enter a panel section, appropriate color coding is maintained for that cabling. In addition, ABB-CE submitted its criteria for identifying associated circuits. The method of identifying associated circuits is in conformance with IEEE 384-1981 and RG 1.75 and readily distinguishes between redundant Class 1E systems, and between Class 1E and non-Class 1E systems. This method is consistent with Position C.11 in RG 1.75 and the recommendations of IEEE 384-1981, and is acceptable. On this basis, DSER Open Item 8.3.1.6-1 is resolved.

8.3.1.7 Physical Separation of Redundant Class 1E Systems

ABB-CE documented that the redundant Class 1E circuits are separated consistent with the guidance of RG 1.75 and recommendations of IEEE 384-1981. The NRC staff had not reviewed the differences between the 1974 and 1981 versions of IEEE 384. Thus, to complete its review, the staff needed ABB-CE to note and justify each difference between the 1974 and 1981 versions of the standard for use on the System 80+ design. (See Section 8.1 of this report.)

In CESSAR-DC Section 8.3.1.4, ABB-CE indicates that the Class 1E redundant circuits and equipment are separated by independent design, distance, or a barrier. The COL applicant will consider hazards in the various areas of the plant in which Class 1E systems are located and will analyze these areas for pipe whips, missiles, and other hazards. Separation or barriers are provided so that damage from hazards does not preclude the performance of a required safety function. The staff lacked sufficient information to determine the amount of separation between the Class 1E circuits and the high-energy piping. In particular, ABB-CE did not state the minimum design objective separation between mutually redundant divisions in areas where redundant circuits are exposed to missile hazards or common hazards, such as flood, pipe whip, or fire. In the DSER, the staff stated that it will need more information to complete its review in this area. This was designated as DSER Open Item 8.3.1.7-1.

In general plant areas of the System 80+ design, cable trays for redundant circuits are separated by a minimum of 0.9m (3 ft) horizontally and 1.5m (5 ft) vertically. Similarly, the Class 1E circuits are separated from non-Class 1E circuits by a minimum of 0.9m (3 ft) horizontally

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and 1.5m (5 ft) vertically. In the areas in which circuits are limited to instrumentation and control functions and those power supply circuits serving the equipment located within the area (non-hazardous area), the separation between the redundant circuits and between Class 1E circuits and non-Class 1E circuits are at a minimum of 0.3m (1 ft) horizontally and 0.9m (3 ft) vertically. Where the plant arrangement precludes maintaining the minimum distance, the circuits requiring separation are run in the enclosed raceways, or barriers are installed between circuits. ABB-CE stated that the COL applicant may use cable enclosed in interlocked armor as a barrier to ensure separation between channels. However, using this type of barrier is inconsistent with RG 1.75, Position C.2, and has not been justified. Therefore, in the DSER, the staff required ABB-CE to revise the CESSAR-DC to comply with the staff's position or justify the design. This was designated as DSER Open Item 8.3.1.7-2.

ABB-CE stated that the minimum distance between the enclosed raceways is 2.54 centimeters (cm) (1-in.). The minimum distance between a conduit and an open raceway is 0.3m (1 ft) horizontally and 0.9m (3 ft) vertically in the non-hazardous areas and 0.9m (3 ft) horizontally and 1.5m (5 ft) vertically in the other areas.

By letter dated July 29, 1991, the staff asked ABB-CE to submit the criteria it used to separate associated circuits from non-Class 1E and other redundant circuits. In the DSER, the staff stated that it needed this information to complete its review in this area. This was designated as DSER Open Item 8.3.1.7-3.

ABB-CE stated that a 0.15m (6-in.) physical separation is maintained between cables of different separation groups inside the control boards. If ABB-CE cannot maintain this separation, it will install barriers between redundant Class 1E wiring.

Electrical penetration assemblies of different Class 1E divisions are physically separated and electrically isolated to maintain the independence of Class 1E circuits. ABB-CE will establish this separation and isolation by locating the redundant penetrations in four quadrants, around the circumference of the containment, each enclosed in a penetration room. The minimum distance between penetrations containing Class 1E circuits or associated circuits and non-Class 1E circuits is 0.9m (3 ft) horizontally and 1.5m (5 ft) vertically.

In the DSER, the staff concluded that the physical independence and separation of Class 1E electrical systems for the System 80+ design was in conformance with the recommendations of IEEE 384-1974, as endorsed by RG 1.75

and acceptable contingent on the resolution of the items discussed above.

In Amendment Q, ABB-CE revised CESSAR-DC Section 8.3.1.4.5.2 to include its minimum separation criteria between (1) mutually redundant Class 1E divisions and between Class 1E circuits and non-Class 1E circuits which are exposed to missile hazards or common hazards, such as flood, pipe whip, or fire, and (2) associated circuits and non-Class 1E and Class 1E circuits. This criterion follows: The areas through which Class 1E and associated circuits are routed and in which equipment is located will be reviewed for the existence of potential hazards. These areas are classified as non-hazard areas, limited-hazard areas, and hazard areas as defined in IEEE 384-1981.

For non-hazard areas, the minimum separation distance between associated circuits and non-Class 1E and other redundant circuits is 0.3m (1 ft) between trays separated horizontally and 0.9m (3 ft) between trays separated vertically, where horizontal separation is measured from the side rail of one tray to the side rail of the adjacent tray, and vertical separation is measured from the bottom of the top tray to the top of the side rail of the tray.

For limited-hazard areas, the minimum separation distance between associated circuits and non-Class 1E and other redundant circuits is 0.9m (3 ft) between trays separated horizontally and 1.5m (5 ft) between trays separated vertically. When the plant arrangements preclude maintaining the minimum separation distance between associated circuits and non-Class 1E and other redundant circuits, these circuits run in enclosed raceways, or barriers are installed between circuits. The minimum distance between these enclosed raceways is 2.5 cm (1 in.).

For hazard areas, those areas of the plant subject to pipe failure hazards, missile hazards, or fire hazards, no Class 1E or associated circuit cables or raceways are routed through the area, except for those cables which terminate at devices or loads within the area. For those cables that must enter the area, the minimum distance between non-Class 1E cable trays and Class 1E or associated circuit cable trays is 0.9m (3 ft) between trays separated horizontally and 1.5m (5 ft) between trays separated vertically. The minimum distances could be increased upon consideration of the cable proximity to hot pipes and upon consideration of the associated hot pipe analysis. Where plant arrangements preclude maintaining this minimum distance, the circuits requiring separation shall be run in enclosed raceways or barriers shall be installed between these circuits. The minimum distance between these enclosed raceways is 2.5 cm (1-in.). Such separation criteria are consistent with IEEE 384-1981 and is acceptable.

With regard to the use of interlocking armor-enclosed cable as a barrier to ensure channel separation, ABB-CE stated that the interlocking armor-enclosed cables are not used as barriers in the System 80+ design. This is consistent with the guidance of RG 1.75, Position C.2, and is acceptable.

On this basis, DSER Open Items 8.3.1.7.1, 8.3.1.7.2, and 8.3.1.7.3 are resolved.

8.3.1.8 Non-Class 1E Loads on Emergency Power Sources

The NRC presently allows licensees to connect non-Class 1E loads to the Class 1E EDGs that supply emergency power to required safety loads. However, the licensee must show that connecting the non-Class 1E loads will not degrade the Class 1E system. ABB-CE indicated that the following non-Class 1E loads; the pressurizer heaters, emergency lighting, RCP seal injection pump, and RCP seal injection pump room ventilation fan are connected to the onsite Class 1E buses. These loads are within the continuous rating of the EDGs. ABB-CE included two breakers in series coordinated with the system bus feed breakers to isolate these loads from the safety buses.

By letter dated July 29, 1991, the staff asked ABB-CE to submit ABB-CE's criteria for isolation devices used to isolate non-Class 1E loads from Class 1E loads. By letter dated January 14, 1992, ABB-CE listed the isolation devices that are used to isolate non-Class 1E loads from Class 1E buses. On this list were control switches, current transformers, and circuit breakers. Although these isolation devices have been approved by IEEE 384-1981, as acceptable isolation devices, the NRC has not endorsed the 1981 version of this standard. Therefore, in the DSER, the staff required ABB-CE to justify the use of these devices as isolation devices on the System 80+ design. This was designated as DSER Open Item 8.3.1.8-1.

In a subsequent submittal, ABB-CE stated that the System 80+ design does not use current transformers as isolation devices. In addition, it was stated that if the COL applicant uses them as isolation devices, they will be qualified per IEEE 384-1981. Circuit breakers and control switches are used as isolation devices. The circuit breakers and control switches used as isolation devices are automatically tripped on an accident signal or they are administratively locked open. The staff contended that since only a few circuits (pressurizer heaters, emergency lighting, RCP seal injection pump and RCP pump room ventilation fan) have been identified as non-Class 1E circuits that need to be isolated from the Class 1E circuits (these circuits are isolated from the Class 1E circuits by two breakers in

series), the other isolation devices should not be listed in the CESSAR-DC.

In Amendment Q, ABB-CE revised CESSAR-DC Section 8.3.1.4.5.1 to delete the list of isolation devices that ABB-CE intended to use to isolate non-Class 1E loads from Class 1E buses. However, ABB-CE has elected to keep the following options: (1) the circuit breakers may be used as isolation devices if they are automatically tripped by an accident signal generated within the same division and (2) if any other device is used, its capability to perform its isolation function will be demonstrated by a qualification test. The qualification test considers the levels and duration of the fault current on the non-Class 1E side. On this basis, the staff concludes that these options for isolating non-Class 1E systems from the safety systems are consistent with the recommendations of RG 1.75 and acceptable subject to the following:

To limit powering of non-Class 1E loads from Class 1E systems, the staff determined that a commitment was required in the design description (Certified Design Material) of electrical systems that states that non-Class 1E circuits connected to Class 1E systems shall be limited to circuits of the pressurizer heaters, emergency lighting, RCP seal injection pump, and RCP seal injection pump room ventilation fan. A similar commitment was made with respect to non-Class 1E loads on the Class 1E 120-V ac power system (see Section 8.3.2 of this report).

ABB-CE included such design commitments in the design description of electrical systems in the certified design material. On this basis, DSER Open Item 8.3.1.8-1 is resolved.

In RG 1.75, Position C.1, the NRC prohibits the use of interrupting devices actuated only by fault current as acceptable devices for isolating non-Class 1E circuits from Class 1E circuits or associated circuits. A circuit breaker can be considered as an isolation device if it is automatically tripped by an accident signal. The staff has also allowed two circuit breakers in series for equipment that must be powered from Class 1E buses. In this case, however, the circuits associated with these loads must be routed as associated circuits to maintain adequate physical separation of these circuits from non-Class 1E or redundant Class 1E circuits.

In the DSER, the staff concluded that placing two circuit breakers in series was an acceptable method for isolating pressurizer heaters, emergency lighting, RCP Seal injection and RCP Seal injection pump room ventilation fan circuits from the safety buses, if the circuits downstream of the isolation devices were treated as associated circuits.

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Therefore, ABB-CE had to affirm that the circuits for the pressurizer heaters and emergency lighting will be treated as associated circuits downstream of the isolation devices. This was designated as DSER Open Item 8.3.1.8-2.

In Amendment Q, ABB-CE revised CESSAR-DC Section 8.3.1.1.2.2 to clarify to the staff that the circuits for the pressurizer heaters, emergency lighting, RCP Seal injection pump and RCP Seal injection pump ventilation fan are treated as associated circuits downstream of the isolation devices in accordance with IEEE 384-1981. This satisfies the staff's requirement and is acceptable. On this basis, DSER Open Item 8.3.1.8-2 is resolved.

8.3.1.9 Cable Derating and Cable Tray Fill

By letter dated July 29, 1991, the staff asked ABB-CE to submit additional information on the criteria for cable derating and cable tray fill for the System 80+ design. By letter dated January 14, 1992, ABB-CE submitted the following criteria for cable derating and cable tray fill.

For the System 80+ design, ABB-CE calculates the maximum ampacities for all power cables by multiplying the appropriate cable manufacturer's cable ampacity rating by 0.7. This produces a 30-percent margin between the rated full load capacity of each power cable and the full load applications. In accordance with Insulated Power Cable Engineering Association (IPCEA) P-46-426, this derating factor applies for 40 °C (104 °F) areas. However, the derating factor for areas warmer than 40 °C (104 °F) is not given. In accordance with IPCEA, the ampacities of cables located in 50 °C (122 °F) areas are derated by a factor of 0.63. In the DSER, the staff stated that ABB-CE must submit additional information on derating factors for cable ampacities in areas other than 40 °C (102 °F). This was designated as DSER Open Item 8.3.1.9-1.

In Amendment Q, ABB-CE stated that the EDG room is subject to the highest ambient temperature within the System 80+ design at a maximum of 50 °C (122 °F). Using highest ambient temperature of the EDG room, ABB-CE calculated the maximum ambient temperature where the 70-percent derating of cables remains adequate. The calculation was performed using Equation 5 in Part ID of the IPCEA standard. The maximum ambient temperature where the 70-percent derating of cables remains adequate was calculated to be 53.36 °C (128 °F). This shows that the 70-percent derating of cables exceeds the IPCEA required derating for temperature throughout the System 80+ design. The 70-percent cable derating factor, based on the maximum temperature of 50 °C (122 °F), covers all areas in the System 80+ design. This satisfies

the staff's concern and is acceptable. On this basis, DSER Open Item 8.3.1.9-1 is resolved.

The cable tray fill criterion for trays containing power cable is as follows. Only a single layer of power cables is routed in any tray, and a separation of one-quarter the diameter of the larger cable is maintained between adjacent power cables within a tray. This criterion is in accordance with IPCEA P-46-426 and is acceptable.

ABB-CE stated that the cable tray fill criterion for trays containing instrumentation and control cables does not exceed the usable cross-sectional area of the tray. This criterion is inconsistent with the recommendations of IPCEA P-46-426. This standard specifies a 40-percent cable fill criterion based on the cross-sectional area or a 50-percent fill based on diameter-squared area, which is equivalent to a 39-percent fill. In the DSER, the staff stated that ABB-CE must revise its cable tray fill criterion for instrumentation and control cables to be consistent with the requirements of IPCEA P-46-426. This was designated as DSER Open Item 8.3.1.9-2.

In a subsequent submittal, ABB-CE stated that the System 80+ cable tray criteria for instrumentation and control cables was stated incorrectly. In Amendment Q, ABB-CE revised CESSAR-DC Section 8.3.1.5.2 to clarify to the staff that the cable fill criteria for those trays containing instrumentation and control cables is that the cross-sectional area of these cables will not exceed 50 percent of the cross-sectional area of the tray. ABB-CE stated that the criterion is in conformance with IEEE 690-1984, "IEEE Standard for the Design and Installation of Cable Systems for Class 1E Circuits in Nuclear Power Generating Stations." This standard recommends a 40-percent to 50-percent fill based on cross-sectional area for instrumentation and control cables. The National Electric Code Section 318-9 (b), also allows for 50-percent fill for cable tray containing instrumentation or control cables. On the basis of the recommendations of these two standards, a 50-percent cable tray fill criterion for instrumentation and control cables is adequate and acceptable. On this basis, DSER Open Item 8.3.1.9-2 is resolved.

8.3.1.10 Criteria for Essential Safety Feature Motors

ABB-CE applied the following design criteria to the ESF pump motors. The criterion for motor size is that the motor will develop sufficient horsepower to accelerate its load under all limiting conditions (maximum expected flow and pressure) in the required time without exceeding the temperature rise rating of the motor. The motors are designed for across-the-line starting. ESF motors, except motor-operated valve (MOV) motors, shall accelerate their

loads within the required time with a starting voltage as low as 75 percent of the rated motor voltage. However, MOV motors shall be capable of accelerating with a starting voltage as low as 80 percent. In the DSER, the staff concluded that ABB-CE needed to address how MOV motors will start when the voltage at the 4.16-kV buses was allowed to drop to 75 percent of the nominal when powered from EDGs. This was designated as DSER Open Item 8.3.1.10-1.

In Amendment Q, ABB-CE revised CESSAR-DC Section 8.3.1.1.8 to clarify that the voltages at the load centers and MCCs which feed MOV motors will always be above 80 percent of the nominal voltage. This is accomplished as follows. Nominal voltage at the load centers and MCCs which feed MOV motors is 480 V. The nominal voltage for the motors is less than for the buses that feed them. Motors fed from the load centers and MCCs have a nominal voltage of 460 V. These are standard voltages for motors for National Electrical Manufacturers Association MG-1, "Motors and Generators." Providing motors with these standard voltages accounts for expected voltage drops in the feeders to the motors.

For the System 80+ design, MOV motors receive power early in the diesel generator loading sequence (at 0.5 seconds for Safety Bus A or B and 4.5 seconds for Safety Bus C or D) prior to load groups with large motors. For the dynamic loading analysis performed for existing nuclear station diesel generators, the lowest voltage dip at these points of the sequence is approximately 94 percent of nominal, or 3,910 V. Thus, the resultant voltage dips at the terminals of the MOV motors will remain well above 80 percent of their nominal voltage of 460 V throughout the sequencing. This corresponds to voltage at the MOV motor terminals to be well above 80 percent of the motor nominal voltage of 460 V.

ABB-CE has stated that voltage studies will be performed for the onsite distribution system by the COL applicant in conformance with Section 3 of BTP PBB-1. This position requires that the voltage levels at the safety-related buses should be optimized for the expected maximum and minimum load conditions. These studies will be verified by test in accordance with Section 4 of BTP (PSB)-1 which requires that the analytical techniques and assumptions used in the voltage analyses performed on the onsite distribution system must be verified by actual measurement. The verification of the analyses by test will demonstrate that the voltage at the terminals of the MOV motors will remain above 80 percent of the nominal voltage throughout out the sequencing. This is acceptable. On this basis, DSER Open Item 8.3.1.10-1 is resolved.

8.3.1.11 Thermal Overload Protection Bypass

MOVs with thermal overload protection devices for the valve motors are typically used in safety systems and their auxiliary supporting systems. Operating experience shows that indiscriminate application of thermal overload protection devices to the motor associated with these valves could result in needless hindrance to successful completion of safety functions. In Position C.1 of RG 1.106, the staff recommends bypassing thermal overload relays during accident conditions. In Position C.2, the staff recommends properly selecting the setpoints for the thermal overloads in a manner that precludes spurious trips.

As originally proposed for the System 80+ design, the thermal overload protection devices on the MOVs were to be bypassed during accident conditions and in force during plant operations. Alarms in the MCR inform the operator of the thermal overload status through the data processing system. This conformed to the guidance of RG 1.106, Position C.1b, and was acceptable. However, the CESSAR-DC did not contain sufficient information on the capability for periodically testing the bypass circuitry. In the DSER, the staff required ABB-CE to submit additional information on the testing of the thermal overload bypass devices. This was designated as DSER Open Item 8.3.1.11-1.

In a subsequent submittal, ABB-CE informed the staff that the thermal overloads are not used to automatically protect Class 1E MOV motors. Although thermal overloads are provided in all phases of the MOV motors, thermal overloads send signals which are used only to indicate alarm and status to the operator via the data processing system displays. Since these thermal overloads do not automatically trip the MOVs under any operating conditions, bypass devices are not used. In Amendment Q, ABB-CE revised CESSAR-DC Section 8.3.1.1.3.2 to delete reference to the periodic testing of the bypass devices since they are not used in System 80+ design. Thus, the design of the thermal overloads used for MOV motors is in conformance with the intent of RG 1.106 with regard to hindering successful completion of safety functions and is acceptable. On this basis, DSER Open Item 8.3.1.11-1 is resolved.

8.3.1.12 Power Lockout to Motor-Operated Valves

The staff asked ABB-CE to submit the following information on safety-related manually controlled electric valves:

- (1) a list of all valves that require power lockout in order to meet the single-failure criterion in the fluid system

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- (2) a description of (a) the design feature for locking out power to these valves, (b) the method for restoring electrical power to the valves from the MCR if the valves need to be repositioned at a later time, and (c) the manner in which the valve position indication satisfies the single-failure criterion.

ABB-CE addressed Item 1 stating that during design certification, it cannot supply a comprehensive list of all System 80+ manually controlled, electrically operated valves for which BTP Instrumentation and Control Branch (ICSB)-18 (PSB) may apply. In the DSER, the staff stated that ABB-CE did not supply sufficient information (for Items 2a, b, and c) for the staff to evaluate the design feature for locking out power to these valves. Therefore, the staff needed additional information to complete its review of both items. This was designated as DSER Open Item 8.3.1.12-1.

In Amendment Q, ABB-CE added a new CESSAR-DC Section 8.3.1.2.11, "Compliance With BTP ICSB-18 (PSB)," to supply the information. ABB-CE has stated that for the System 80+ design BTP ICSB-18 (PSB) applies only to safety injection tank (SIT) discharge isolation MOVs, SIT vent solenoid valves, and the RCP seal injection high-pressure (HP) seal cooler isolation valves. BTP-18 applies to no other valves. The power lockout to these valves is achieved as follows.

For the SIT discharge and the RCP seal injection HP seal cooler isolation valves, the electrical motive power is removed by racking out the breaker at the MCC. For the SIT vent solenoid valve, the electrical power is removed by removing fuses at the ESF-CCS loop controller enclosure. Administrative procedures and local locks preclude unauthorized re-energization of these components. The electrical power is restored by racking back in the MOV breaker at the MCC and reinstalling the solenoid valve fuses at the ESF-CCS loop controller enclosures.

For the SIT discharge valves, redundant valve position indication is provided in the MCR using redundant and diverse sets of open/close position limit switches (one set on the valve stem and one set on its motor gearbox). The RCP seal injection HP seal cooler isolation valves are non-safety-related and are not classified as "active valves." For these valves, redundant valve position status is provided in the MCR via the process CCS and data processing system display devices. For the solenoid valves, a single set of valve stem limit switches for open/close position is provided due to physical valve design limitations. Redundancy and diversity are achieved by tank pressure instrumentation. Inadvertent opening of the SIT solenoid vent valve will quickly depressurize the SIT, causing an alarm in the MCR.

The methods of power lockout, and the valve position indication are designed in accordance with BTP ICSB-18 with regard to application of the single failure criterion to manually-controlled electrically-operated valves and are, therefore, acceptable. On this basis, DSER Open Item 8.3.1.12-1 is resolved.

8.3.1.13 Containment Electrical Penetrations

GDC 50 OF 10 CFR Part 50, Appendix A, requires that the reactor containment structure, including penetrations, be designed so that the containment structure can, without exceeding the design leakage rate, accommodate the calculated pressure, temperature, and other environmental conditions resulting from any loss-of-coolant accident.

To meet the applicable requirements of GDC 50, the recommendations of IEEE 317-1982, and the guidance of RG 1.63, the containment electrical penetration assemblies is designed to withstand, without any loss of mechanical integrity, all possible short-circuit conditions, including the maximum available fault current and time conditions that could occur given single random failure of a circuit overload protection device. The circuit overload protection system should conform to the criteria of IEEE 603-1980, for testing and independence.

All electrical circuits for the System 80+ design which pass through containment penetration assemblies have primary and backup protective devices, unless the maximum fault current available in the circuit is less than the continuous rating of the penetration assembly. ABB-CE indicated that the resistance temperature detectors, thermocouples, transducers, and annunciator circuits have maximum currents below the continuous rating of the penetration assembly. The staff considers this to mean that these circuits will not have redundant protective devices. To accept this design, the staff will need information on the qualification tests performed to demonstrate that these current-limiting devices would limit the maximum current available to less than the continuous current rating of the penetration assembly. Control transformers or other devices that could lose their fault current-limiting capacity under fault or overload conditions may not be used as current-limiting devices.

ABB-CE stated that the primary and secondary protective devices for circuits passing through containment penetration assemblies are coordinated to ensure that the circuit is interrupted before it reaches the maximum time-current capability of the penetration assembly. The primary and secondary protection devices are consistent with the requirements of IEEE 603-1980 for independence and testing.

Having evaluated these criteria, the staff concludes that the System 80+ design provides independent primary and backup fault protection for each penetration conductor (unless the maximum overcurrent of the circuits does not exceed the continuous rating of the containment electrical penetration assembly) to preclude a single failure from impairing the integrity of the containment electrical penetration. This is consistent with the guidance of RG 1.63 and IEEE 317-1982 and is acceptable.

By letter dated July 29, 1991, the staff asked ABB-CE to submit coordinated fault-current-versus-time curves for each type of cable that penetrates the primary containment. The curves for each cable must show the relationship between the capability of the electrical penetration to carry fault current and the protection provided by the primary overcurrent protective device.

By letter dated January 14, 1992, ABB-CE stated that to provide such curves for each type of cable that passes through containment penetration, it would need as-procured information, which is beyond the scope of design certification. ABB-CE will perform a preliminary fault current study for all buses under bounding conditions to ensure that all buses and interrupting devices are specified to have adequate capabilities to withstand and interrupt faults. The COL applicant will perform a final fault current study upon determining the actual transformer, cable, and bus impedances.

In the DSER, the staff concluded that ABB-CE should submit appropriate inspections, tests, analyses, and acceptance criteria (ITAACs) for the containment electrical penetrations design for the System 80+ design. This was designated as DSER Open Item 8.3.1.13-1.

DSER Open Item 8.3.1.13-1 was related to the design description and the ITAACs for the containment electrical penetration design. The adequacy and acceptability of the System 80+ design description and ITAACs are evaluated in Chapter 14 of this report. On that basis, DSER Open Item 8.3.1.13-1 is resolved.

8.3.1.14 Adequacy of Station Electric Distribution System Voltages

Events at the Millstone Nuclear Power Station have shown that sustained low- grid voltage conditions can adversely affect the Class 1E loads when the Class 1E buses are connected to offsite power. The low-voltage conditions will not be detected by the loss-of-voltage-pickup setting, which is generally in the range of 0.7 per unit voltage or less. Therefore, the staff required that in addition to the undervoltage scheme to detect LOOP at the Class 1E buses, a second level of undervoltage (degraded) protection

with time delay to protect the Class 1E equipment should also be provided.

In the System 80+ design, ABB-CE includes, in addition to the undervoltage relays for each safety bus (A, B, C, D) and each permanent non-safety bus (X, Y), a second level of undervoltage protection for Class 1E and permanent non-safety buses. These schemes are designed in accordance with the recommendations of IEEE 741-1986 and BTP PSB-1. The COL applicant will select two separate time delays for the second level of undervoltage protection. A sustained loss of voltage on either of the safety buses will cause the affected safety buses to isolate themselves from the offsite source and to start the divisional EDG. Since each EDG feeds two safety buses for the division (A and C for Division I, B, and D for Division II), it was not clear if the sustained undervoltage on any bus will start the EDG. In CESSAR-DC Section 8.3.1.1.6, ABB-CE indicated that the undervoltage protection schemes use coincident logic of 2-out-of-3 to avoid spurious trips of the offsite power sources. ABB-CE did not indicate how three relays are distributed among the two buses for each division. In the DSER, the staff stated that to fully understand this design, the staff needed additional information in the following areas:

- (1) the number of undervoltage relays and second level of undervoltage relays provided for each safety bus and the coincident logic provided for these relays
- (2) the number of undervoltage relays and second level of undervoltage relays provided for each permanent non-safety bus and the coincident logic provided for these relays
- (3) the relay setpoints for both the schemes and the associated time delays
- (4) the voltage studies performed on the Class 1E onsite power distribution system.

This was designated as DSER Open Item 8.3.1.14-1.

In Amendment Q, ABB-CE revised CESSAR-DC Section 8.3.1.1.6 to include the additional information. Class 1E divisions and permanent non-safety buses have separate bus voltage monitoring and protection schemes for degraded voltage and loss of voltage conditions, respectively. These schemes are designed in conformance with the recommendations of IEEE 741-1986 and BTP PSB-1. These protection schemes are designed as follows:

There are a total of three first-level undervoltage relays to detect loss of voltage and three second-level undervoltage relays to detect degraded voltage on each of the four safety

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buses. The first level of undervoltage relays comprises definite time relays with built-in time delays of approximately 10 cycles to ride out power system transients. The initiation of any two out of the three first-level undervoltage relays within a single switchgear starts the diesel generator and, if voltage has not returned by the time the EDG is up to speed and voltage, (1) the main bus feeder breakers for both Class 1E switchgears in the division are tripped, (2) loads are shed, (3) EDG output breakers close on both of the division's switchgear (each of the division's Class 1E switchgear group's EDG has a breaker (a third breaker at the generator itself is normally closed and used for maintenance purposes), and (4) sequencing of safety loads begins.

The initiation of any two of the second-level instantaneous undervoltage relays within a single switchgear starts a timer to establish existence of a sustained undervoltage. This timer will be set to allow the largest motor to accelerate before initiating an alarm in the MCR. The subsequent occurrence of an ESF signal immediately separates the Class 1E switchgear from the offsite power system. If no ESF signal is present and the operator has not restored voltages after a second time delay, the Class 1E switchgear is immediately separated from the offsite power system. The second time delay is of such a limited duration that the permanently connected Class 1E loads will not be damaged.

The dropout for the second level of undervoltage relays is set at a level above the minimum voltage that allows operation of Class 1E loads with the worst-case lineup and minimum switchyard voltage. The associated first time delay is set to establish existence of a sustained undervoltage (something longer than a motor start — approximately 10 seconds).

There are three first-level and three second-level undervoltage relays on each permanent non-safety bus. The first level of undervoltage protection is a set of definite time undervoltage relays with inherent time delays of approximately 10 cycles to ride out power system transients. The initiation of any two of the first-level undervoltage relays within a single switchgear initiates a residual voltage transfer to offsite power by tripping the switchgear breaker. The transfer is completed when an instantaneous undervoltage relay indicates the bus voltage has decayed to 25 percent. If the voltage has not recovered after a second time delay of approximately 10 seconds, the CTG (AAC source) starts. If voltage has not recovered after the CTG is up to speed (approximately 2 minutes), then (1) the main feeder breakers for both permanent non-safety switchgears are tripped, (2) the load shed occurs, (3) the CTG breakers close on both switchgear, and (4) sequencing of loads begins.

The dropout for the second level of undervoltage protection for the permanent non-safety-related distribution system is set at a level above the minimum voltage that allows proper operation of permanent non-safety-related loads with the worst-case lineup and minimum switchyard voltage. The associated time delay is set to establish existence of a sustained undervoltage (something longer than a motor start — approximately 10 seconds). The second time delay is limited so that the connected equipment is not exposed to prolonged voltage below the equipment rating.

These relay settings and time delays of all levels of undervoltage protection are established on the basis of the voltage studies that will be performed by the COL applicant in conformance with BTP PSB-1. The results will be verified by a test as described in Position 4 of BTP PSB-1.

The design of the station electrical distribution system voltages for the System 80+ is consistent with the guidelines of BTP PSB-1 and IEEE 741-1986 and is acceptable. On this basis, DSER Open Item 8.3.1.14-1 is resolved.

8.3.1.15 Protective Relaying

Upon reviewing the applications of protective relays in Class 1E electrical system equipment, the staff found that the drift in the relay trip setpoints of conventional relays has caused pump motors in redundant safety-related systems to trip prematurely when the safety systems were required to be operative. The staff recognizes that feeders and equipment should be properly protected against permanent faults. However, spurious trips of protective relays should not cause redundant safety systems to be totally unavailable. The primary function of the electrical distribution system is to supply reliable power to safety-related equipment.

Protective relays used throughout the electrical systems for the System 80+ design employ solid-state rather than electromechanical components. ABB-CE stated that solid-state relays of current technology, which are "primarily microprocessor or integrated circuit based," are typically not susceptible to setpoint drift. In the DSER, the staff agreed that the typical applications of solid-state relays are more reliable and are not susceptible to setpoint drift. However, the staff stated that it needed additional information in the following areas to complete its review:

- (1) the process for periodically verifying and calibrating the relay trip setpoint to ensure it performs its safety function when required
- (2) the typical design margin between the relay setpoint and the overload rating of the loads

- (3) a list of systems that use microprocessors or integrated circuits

This was designated as DSER Open Item 8.3.1.15-1.

In Amendment Q, ABB-CE stated that the COL applicant will provide the process for periodically verifying and calibrating the relay trip setpoint to ensure that it performs its function when required. ABB-CE has included this as a COL action item in CESSAR-DC Section 8.1.4.5. All relays will be tested and calibrated every refueling outage according to the relay manufacturer's detailed instructions which outline acceptance criteria, test connections, and test procedures.

With regard to Item 2 above, ABB-CE stated that the pickup of the overload relays used for protection of motors is set in accordance with IEEE 242-1986, "IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems." This criterion allows the relay to be set so that its trip curve lies between the motor starting curve and the motor thermal damage curve. The overload protection for transformers is also consistent with IEEE 242-1986. The overload protection for transformers is set to pick up at a value greater than 125 percent of the maximum (forced cooling) rating of the transformer. The overload protection is set with a margin of 15 percent or more between the relay setting and the thermal damage curve of the transformer to ensure operation of the relay before the transformer sustains damage.

In response to Item 3 above, ABB-CE stated that nearly all systems in the System 80+ design use microprocessor or integrated circuit-based devices. ABB-CE has also listed equipment that uses microprocessor or integrated-based devices, as requested by the staff.

The protective relaying setpoint for the System 80+ design is periodically verified and calibrated and these protective devices are periodically subjected to a functional test to ensure that they perform their safety function when required. The proposed design satisfies the protection requirements of GDC 17 and is acceptable. On this basis, DSER Open Item 8.3.1.15-1 is resolved.

8.3.1.16 Breaker Coordination

The staff was concerned that improper breaker coordination could cause the loss of needed equipment during fault conditions. The staff requires that the load breakers be coordinated with the bus feeder breakers to prevent the bus feeder breaker from tripping, since tripping of a bus feeder breaker could cause the loss of multiple safety equipment.

ABB-CE stated that it will perform circuit breaker coordination studies to verify that the circuit breaker closest to the fault opens before upstream breakers. In the DSER, the staff stated that this solution was acceptable if ABB-CE also developed ITAACs for verifying these design commitments. This was designated as DSER Open Item 8.3.1.16-1.

DSER Open Item 8.3.1.16-1 was related to the description and the ITAAC for the breaker coordination design. The adequacy and acceptability of the ABB-CE design description and ITAAC are evaluated in Chapter 14 of this report. On this basis, DSER Open Item 8.3.1.16-1 is resolved.

8.3.1.17 Equipment Fault Current Interrupting Capability

The staff conducted electrical distribution system function inspections at various operating plants and found that some switchgear serving ESF circuits were not of sufficient size to operate properly under all possible short-circuit conditions. Certain circuit breakers had inadequate fault-current interrupting capability for the duties to which they were assigned. If a circuit breaker has an inadequate fault-current interrupting capability, it could cause an unacceptable level of equipment loss during fault conditions.

ABB-CE stated it would perform a preliminary fault-current study for all buses under bounding conditions to ensure that all buses and interrupting devices are specified to have adequate capabilities to withstand and interrupt faults. ABB-CE will perform a final fault-current study once the actual transformer, cable, and bus impedances are known. In determining the appropriate size of the switchgear, ABB-CE will consider the fault-current contribution from the EDG when it is operating in parallel with the offsite power. In the DSER, the staff concluded that this solution satisfied the staff's concern and was acceptable if ABB-CE includes ITAAC which include these design commitments as acceptance criteria. This was designated as DSER Open Item 8.3.1.17-1.

DSER Open Item 8.3.1.17-1 was related to design description and the ITAACs for the sizing of the switchgears serving ESF circuits. The adequacy and acceptability of the System 80+ design description and ITAACs are evaluated in Chapter 14 of this report. On this basis, DSER Open Item 8.3.1.17-1 is resolved.

8.3.1.18 Submergence

By letter dated January 29, 1992, the staff asked ABB-CE to identify all electrical equipment, both safety-related and non-safety-related, that may become submerged as a result

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of a LOCA. The staff required that ABB-CE to analyze all equipment that is not qualified for service in such an environment. In performing this analysis, ABB-CE would (1) determine the safety significance of the failure of this electrical equipment, such as by spurious actuation or loss of actuation function as a result of flooding, and (2) determine the effects of submergence on Class 1E electrical power sources serving this equipment.

By letter dated January 14, 1992, ABB-CE stated that in the System 80+ design, six dc-operated valves are inside the containment and are used only under severe accident scenarios. These valves are qualified to operate in flooded conditions. Other electrical equipment is purposely located away from areas of possible flooding. This information indicates that all other electrical equipment will be located above the flood level.

In the System 80+ design, divisions are physically separated and electrical and mechanical equipment is segregated to protect the electrical equipment from flooding that a LOCA could cause outside the containment. In the lowest elevation susceptible to flooding in the System 80+ design, the COL applicant will use these principles, particularly the design segregation of electrical equipment from mechanical equipment, to prevent the flooding resulting from a failure of a mechanical system from affecting electrical equipment. At higher elevations, the COL applicant will separate divisions and will install flood doors and curbs to protect electrical equipment.

In the DSER, the staff stated that this information did not indicate whether the electrical equipment outside the containment will be submerged. Therefore, ABB-CE needed to confirm that all electrical equipment that could be submerged will be qualified. This was designated as DSER Open Item 8.3.1.18-1.

In Amendment Q, ABB-CE confirmed that all electrical equipment for which operation in a submerged condition is credible and is required for safely shutting down the plant or mitigating the consequences of an accident, is qualified for such submerged operation. Where electrical equipment operability during submergence is required, qualification is demonstrated by type test and/or analysis. On this basis, the staff concludes that all Class 1E equipment subject to submergence is either designed and qualified to perform its required safety function without failing while submerged, or is appropriately protected from submergence. This meets the protection requirements of GDC 4 and is acceptable. On this basis, DSER Open Item 8.3.1.18-1 is resolved.

8.3.1.19 Security System Power Supply

In a request for additional information (500.20), the staff asked whether security system loads would be included among the permanent non-safety-related loads as then specified in EPRI's Advanced Light-Water Reactor (ALWR) URD. (Subsequently, by letter dated January 24, 1992, EPRI revised Chapter 9 of the URD to require instead a dedicated security power generator and an uninterruptible power supply (UPS).)

By letter of December 17, 1991, ABB-CE replied that onsite power for security alarm annunciators and stationary communications equipment is powered from an UPS with dedicated batteries, which in turn is kept charged by the permanent non-safety buses. The permanent non-safety buses will be connected to normal power, backed up by the CTG (AAC source). In Amendment Q, ABB-CE added the following statement to CESSAR-DC Section 8.3.1.1.5: "The AAC facility is located within the plant protected area, outside of the turbine missile impact zone, and is surrounded by a perimeter security fence and intrusion alarm system. The security fence is locked, and monitored in the site security office." This resolution is acceptable because it satisfies the requirement of 10 CFR 73.55(e)(1). This was designated as DSER Confirmatory Item 8.3.1.19-1. On the basis of this evaluation, DSER Confirmatory Item 8.3.1.19-1 is resolved.

8.3.1.20 Cable Penetrations Through Security Barriers

Penetration of cables through fire barriers is discussed in CESSAR-DC Section 8.3.1.6; however, the CESSAR-DC document lacked a similar discussion of cable penetrations through security barriers. In RG 5.65, the NRC provides guidance on physical barriers at cable tray penetrations of vital area boundaries. In Underwriters Laboratories Standard 752 and NUREG/CR-0543, "Central Alarm Station and Secondary Alarm Station Planning Document," the industry and the NRC provide guidance for cable penetrations of those bullet-resistant security barriers required for personnel protection, such as the control room security boundary. Cable penetrations of security barriers was identified as DSER Open Item 8.3.1.20-1.

By Amendment S, ABB-CE added the following statements to CESSAR-DC Section 8.3.1.4.5.1. "Tray penetrations through security barriers and all other security considerations are designed as stated in Appendix 13A, Section 2.2., Item F, Industrial Security." In Appendix 13A, ABB-CE added the following statement, "Penetrations to vital areas for piping, electrical power, instrument and controls, support systems, [heating, ventilation, and air conditioning] shall be constructed so as to prevent unde-

ected personnel ingress." This resolution is acceptable. On this basis, DSER Open Item 3.1.20-1 is resolved.

8.3.2 Direct Current Power System

The direct current (dc) power system supplies (1) control power to the alternating current (ac) offsite and onsite emergency power systems as required, (2) power to the four inverters for the Class 1E vital instrumentation and control ac power systems, and (3) motive and control power to selected safety-related equipment. The staff's objective is to confirm that the dc power system is testable; meets the single-failure criterion; has the required redundancy; and has the capacity, capability, and reliability to supply power to all required safety loads in accordance with GDC 2, 4, 17, and 18.

The Class 1E dc power system for the System 80+ design consists of four independent and physically separated dc subsystems that supply instrumentation and control Channels A, B, C, and D. Each subsystem has a battery, a battery charger, a dc distribution center, and an associated panel board. Each division also has an additional battery, battery charger, dc distribution center, and an associated panel board to supply control power to the divisional ac 4.16-kV switchgear and 480-V load centers. Class 1E dc systems are designated as Division I and Division II.

During normal operation, the battery charger supplies the load with the battery floating on the system. The battery chargers of load group Channels A and C, and Division I are powered from Division I of the Class 1E ac bus. Division II of the Class 1E ac bus powers the battery chargers of load group Channels B and D and Division II. Upon loss of the ac power, the entire dc load is powered from the battery until the EDG restores ac power. Each Class 1E battery has sufficient capacity to supply a complete set of channel loads and a set of division battery loads for a minimum of 2 hours. The batteries are of sufficient size to permit operation of the SBO coping systems for 8 hours, assuming manual load shedding and load management programs.

Each battery charger has enough capacity to recharge the battery from its design minimum charge state to the fully charged state within 8 hours while simultaneously supplying the largest combined demand of various steady-state loads irrespective of the status of the plant during which these demands occur. This capacity is consistent with the guidance of RG 1.32. ABB-CE shall also provide for the connection of the Division I dc bus to feed safety Channel A or C and the connection of the Division II dc bus to safety Channel B or D through normally open breakers. This provision will allow one battery to be inoperable without requiring the COL applicant to violate the limiting

condition for operation. If the battery is removed from the bus, the two connected channels would have one battery and two chargers in operation. This mode of operation is acceptable if the short-circuit rating of the dc distribution buses and breakers of each safety channel is sufficient to withstand the short-circuit contribution from both chargers and a division battery.

In the DSER, the staff stated that the proposed design of the battery chargers was acceptable if ABB-CE confirmed (1) that the dc switchgears of all channel batteries and division batteries are rated to withstand the short-circuit contribution from two chargers and a battery, and (2) that each Class 1E battery charger has sufficient capacity to power both its own group loads and the bus loads it will power if any one of the Class 1E batteries is inoperable. This was designated as DSER Confirmatory Item 8.3.2-1.

In Amendment Q, ABB-CE confirmed that the dc switchgear of all channel batteries and division batteries are rated to withstand the short-circuit contribution from two chargers and a battery. ABB-CE also confirmed that each Class 1E battery charger has sufficient capacity to power both its own group loads and the bus loads it will power if any one of the Class 1E batteries is inoperable. This satisfies the staff's concern and is acceptable. On this basis, DSER Confirmatory Item 8.3.2-1 is resolved.

In accordance with the requirements of GDC 2 which require that Class 1E systems, equipment, and components be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions, the Class 1E batteries, chargers, and dc switchgear are located in a seismic Category I control building which protects the equipment from the effects of tornadoes, missiles, and floods. ABB-CE designated the Class 1E dc power systems as quality assurance (QA) designation Class 1E and designated them to be seismically and environmentally qualified to withstand the environmental conditions associated with normal operation and postulated accidents. This is also in accordance with the requirements of GDC 4 which requires that Class 1E systems, equipment and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents.

Each of the Class 1E dc subsystem batteries, chargers, switchgears, and distribution systems is physically separate and independent from its redundant counterparts. The Class 1E dc systems have sufficient capacity, capability, independence, redundancy, and testability to ensure that safety functions will be performed if a single failure occurs. This is consistent with GDC 17 which requires

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that the Class 1E systems have sufficient independence and redundancy to perform their safety functions assuming a single failures. ABB-CE designed each Class 1E dc system to permit the COL applicant to inspect and test the system for operability and continuity during plant operation and shutdown conditions. This is consistent with the requirements of GDC 18 with regard to periodic inspection and testing of electric power systems important to safety.

By letter dated July 29, 1991, the staff asked ABB-CE to list all non-Class 1E loads that are connected to the Class 1E batteries and describe how these non-Class 1E loads are isolated from the Class 1E batteries. ABB-CE responded on January 14, 1992, that all loads on the Class 1E batteries, both Class 1E and non-Class 1E, are required for a design-basis accident, LOOP, SBO, or severe accident, and satisfy the separation guidelines of RG 1.75. In the DSER, the staff stated that ABB-CE did not, as requested by the staff, list the non-Class 1E loads that are connected to the Class 1E batteries and did not describe the process for isolating these loads from the Class 1E batteries. Therefore, the staff needed the following additional information to complete its review in this area:

- (1) list all non-Class 1E circuits that are connected to the Class 1E batteries and describe the manner in which these non-Class 1E loads are isolated from the Class 1E batteries
- (2) list the non-Class 1E loads that are connected to the Class 1E batteries and that must be operable during design-basis accidents

This was designated as DSER Open Item 8.3.2-1.

In Amendment Q, ABB-CE listed containment equipment hatch trolley circuits as non-Class 1E loads that will be connected to the Class 1E batteries and hydrogen igniter, reactor cavity flood valves, and holdup volume flood valve circuits as non-Class 1E loads that will be connected to the ac instrumentation and control power system. ABB-CE will provide two breakers in series for isolating these loads from the safety buses. The cabling downstream of the isolation devices will be treated as associated circuits to keep adequate separation of these circuits from other non-Class 1E or redundant Class 1E circuits. This meets the separation recommendation of RG 1.75 and is acceptable. To limit powering of non-Class 1E loads from Class 1E systems, the staff has determined that a commitment was required in the design description of electrical systems that states that non-Class 1E circuits connected to Class 1E systems shall be limited to containment hatch trolley, hydrogen igniter, holdup volume flood valve, and reactor cavity flood valve circuits.

ABB-CE included such a design commitment in the design description of electrical systems in the certified design material. Based on this, DSER Open Item 8.3.2.1 is resolved.

To ensure that the Class 1E dc power systems continue to operate, ABB-CE will install local and MCR indication and alarms to monitor the status of the batteries and battery chargers for the System 80+ design. By letter dated July 29, 1991, the staff informed ABB-CE that the list of alarms and indication in CESSAR-DC Section 8.3.2.1.2.1.5 was inadequate and required additional alarms and indications in the following areas:

- (1) specify that the battery ammeter must be able to monitor the battery charge and discharge currents. This is needed to know the battery status for all plant states.
- (2) add a battery discharge alarm to the list of required alarms. This alarm is needed for alerting the operator to a battery discharge condition before the battery loses a significant amount of capacity
- (3) add alarms for "battery circuit breaker open" and "battery charger input breaker open."
- (4) specify in the requirement for ground fault detectors that sensitive ground-monitoring alarms with high source impedance shall be installed. This is necessary so that the alarm can detect high-resistance grounds on the dc system without creating another ground on the system

By letter dated January 14, 1992, ABB-CE revised CESSAR-DC Section 8.3.2.1.2.1.5 to include these items. This is acceptable. In the DSER, the staff stated that it needed the following additional information to complete its safety review of the dc power system:

- (1) indicate for each parameter listed in CESSAR-DC Section 8.3.2.1.2.1.5, whether it alarms, or indicates or does both
- (2) list the indication/alarms located at the battery and battery charger local panels
- (3) justify not including a dc voltmeter to monitor bus voltage
- (4) justify not including a dc bus overvoltage alarm

This was designated as DSER Open Item 8.3.2-2.

In Amendment Q, ABB-CE revised CESSAR-DC Section 8.3.2.1.2.1.5, "125 V dc and 120 V ac Vital Instrumentation and Control Power System Status Information," to include the information requested. The following parameters have an alarm only in the MCR:

- battery charger output voltage low
- battery charger output voltage high
- loss of ac input to battery charger
- vital 125-V dc panelboard undervoltage
- battery positive or negative leg ground
- inverter 125-V dc input failure
- inverter output voltage low
- inverter alternate source abnormal (voltage or frequency)
- 120-V ac inverter panelboard undervoltage
- battery discharge alarm

The following parameters have indication only in the MCR.

- battery current (charge and discharge)
- 125-V dc switchgear voltage

The following parameters have both an alarm and indication in the MCR:

- battery charger output circuit breaker open
- distribution center main circuit breaker open
- battery voltage (high and low)
- inverter manual bypass switch in alternate source position
- static inverter manual bypass switch position
- 120-V ac regulated distribution center breaker status
- battery circuit breaker open
- vital distribution center tie breaker closed

The following indications are located at the battery and battery charger control panels:

- dc voltage
- dc current
- battery charger ac supply on

The following alarms are located at the battery and battery charger control panels:

- ac failure
- low dc voltage
- high dc voltage
- positive leg ground fault
- negative leg ground fault
- battery discharge

ABB-CE will provide a dc voltmeter at the charger local panel and in the MCR as required by the staff.

On the basis of such status indication and alarms for the Class 1E dc power system, the indications and alarms cited above, and augmented by the periodic test and surveillance requirements, provide reasonable assurance that the Class 1E dc power system will perform its safety function as required. On this basis, DSER Open Item 8.3.2-2 is resolved.

8.3.2.1 Loss of an AC or DC Bus

In a meeting with ABB-CE on April 30, 1992, the staff asked ABB-CE to perform a failure modes and effects analysis to ensure that failure of any ac or dc bus will not result in a plant transient and simultaneously cause the loss of single-failure protection in any safety-related system.

ABB-CE subsequently submitted information on the modified electrical distribution system for the System 80+ design. In the DSER, the staff concluded that ABB-CE had not adequately addressed this item. This was designated as DSER Open Item 8.3.2.1-1.

Subsequently, ABB-CE illustrated that the System 80+ design ensures that failure or loss of any ac or dc bus does not result in a plant transient and simultaneously cause the loss of single-failure protection in any safety-related system. The System 80+ design either prevents a plant transient, or if a transient is not preventable, the design may take the most conservative single failure following or in conjunction with the transient, and still have sufficient safety-related equipment and systems to safely shut down the reactor and bring the plant to safe shutdown. This is accomplished as follows:

For failure of any of the 13.8-kV non-safety buses, a unit will trip. For this transient, both safety divisions (I and II) remain available. A single failure of an entire safety division still leaves the unit with an intact safety division to safely shut down the reactor down and bring the unit to a safe-shutdown condition.

The same analysis applies to the 4.16-kV non-safety and permanent non-safety buses. A failure of either a 4.16-kV non-safety or a permanent non-safety bus causes a plant trip. Assuming a unit does trip, a single failure of either safety division may be taken and still leave the unit with an intact safety division to safely shut the reactor down and bring the unit to a safe-shutdown condition.

For the 4.16-kV safety buses, loss of any one safety bus (A, B, C, D) will not cause a plant transient, since sufficient redundancy and diversity exist in the System

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80+ design. In the most conservative 4.16-kV safety bus failure, the component cooling water and station service water pumps servicing the divisional component cooling water heat loads were being powered from the failed bus prior to the failure. Following the failure, the operator must align the second division component cooling water and station service water pumps to continue to serve divisional component cooling water heat loads. The System 80+ design allows such pump alignment to be accomplished within 30 minutes to maintain the unit on line and thus avoid a plant transient.

For the 125-V dc vital power system buses, sufficient component redundancy and diversity exist so that no transient will occur should any one Class 1E dc bus cease to function.

For the 120-V ac vital power system buses, sufficient component redundancy exists so that no transient will occur should any one 120-V ac Class 1E bus cease to function.

On this basis, the Class 1E power system for the System 80+ design supports the performance of all safety functions required for a design-basis accident in the presence of failure of an ac or dc bus. Further, a failure of a dc or ac bus will not result in a plant transient and simultaneously cause the loss of single-failure protection in any safety system. This is consistent with the single-failure criterion requirements of IEEE 308-1980, and is acceptable. On this basis, DSER Open Item 8.3.2.1-1 is resolved.

8.3.2.2 Non-Class 1E DC Power Systems

ABB-CE includes two non-Class 1E 125-V batteries, battery chargers, and distribution systems to supply dc power to the non-Class 1E instrumentation and controls. A separate non-Class 1E 125-V battery, battery charger, and distribution panel supply the dc power to start and operate the AAC source. Two other 250-V batteries, battery chargers, and distribution panels supply 250-V dc power for high-inrush dc loads.

8.3.2.3 Vital AC Instrumentation and Control Power Supply

Four redundant and independent divisions of 120-V Class 1E vital ac instrumentation and control power subsystems supply power to the four channels (A, B, C, and D) of the reactor trip system and the ESFAS. The 120-V ac vital instrumentation and control power system consists of four separate and independent 120-V ac power panels, each powered from a 125-V dc load group distribution center through a 125-V dc-120-V ac static inverter. This power

supply system produces an output frequency of 60 ± 0.5 Hz and voltage regulation to within ± 2 percent. An autostatic switch for each vital bus instantly transfers the load from the output of the inverter to a regulating transformer which is fed from a Class 1E 480-V ac MCC. Each vital bus can also receive backup power from the regulating transformer through a manual make-before-break bypass switch during periods in which the inverter is out of service for maintenance.

The design includes two additional 120-V ac power panels which are powered from the independent Division I and II 125-V dc battery and battery charger described in Section 8.3.2 (above). Four 120-V ac power panels (two for each division) are powered from the ac-regulated Class 1E 480-V/120-V ac regulated transformer. In the DSER, the staff stated that ABB-CE did not indicate which loads would be powered from these panels. Therefore, the staff needed the following information to complete its review of the 120-V ac vital instrumentation and control systems:

- (1) a list of the loads powered from vital Division I and II 120-V ac distribution panels and ESF-component control system panels
- (2) the normal feed to ESF-CCS panels and the conditions under which these panels may be powered from the inverters

This was designated as DSER Open Item 8.3.2.3-1.

In Amendment Q, ABB-CE provided loads powered from the ESF-CCS panels. The ESF-CCS provides normal control of safety components, as well as control and actuation of ESF systems. The ESF-CCS is the central controlling system for safety-related plant components. Examples of ESF signals controlled by ESF-CCS follow:

- containment isolation actuation signal.
- containment spray actuation signal
- main steam isolation signal
- SIAS
- emergency feedwater (EFW) actuation signal

The ESF-CCS panel has a normal feed and an alternate feed. One feed is from the Class 1E vital I&C channel power panel, and the other feed is from the same Class 1E channel 480-V/120-V ac regulated transformer. The ESF-CCS panel automatically transfers internally to its alternate source if the inverter is inoperable or if the inverter is to be removed from service for maintenance.

This satisfies the staff's concern and is acceptable. On this basis, DSER Open Item 8.3.2.3-1 is resolved.

8.3.2.4 Non-Class 1E AC Instrumentation and Control Systems

The design includes two separate non-Class 1E 208-V/120-V ac control systems consisting of 125-V dc power to 208-V/120-V ac inverters, static and manual bypass transfer switches, distribution centers, and panels. Two 125-V dc to 120-V ac inverters supply power from 125-V dc control power to separate 208-V/120-V ac panel boards. A non-safety-related power transformer supplies backup power to these panels. This system consists of two subsystems, each supplying non-interruptible 120-V ac power to non-Class 1E instrumentation and controls, and 208-V/120-V ac power to the data processing system and security lighting systems.

8.4 TMI Action Plan Requirements

Item II.E.3.1, "Pressurizer Heater Power Supply"

This item requires that emergency power be available to a minimum number of pressurizer heaters to ensure that natural circulation can be maintained in the RCS if offsite power is lost.

Although no credit is taken for pressurizer heaters to maintain natural circulation if offsite power is lost, the System 80+ design includes two backup pressurizer heater groups, each rated at 200 kW. These heaters are connected to separate 480-V Class 1E buses that are energized from separate and independent EDGs upon the LOOP. Because the heaters are not Class 1E, they are connected to the Class 1E buses through two breakers in series. The criteria for this power supply for pressurizer heaters are acceptable because they are consistent with NUREG-0737, "Clarification of TMI Action Plan Requirements," requirements.

Item II.G.1, "Power Supplies for Pressurizer Relief Valves, Block Valves, and Pressurizer Level Indicators"

This item requires that emergency power be available to pressurizer relief and block valves and pressurizer level indication in the event of LOOP. The System 80+ design does not incorporate pressurizer relief and block valves. The safety depressurization system (SDS) performs a rapid depressurization of the RCS to enable the operator to feed and bleed the RCS during beyond-design-basis events. Since this system is designated Class 1E, the systems and components, including the pressurizer level indication and SDS valves, are powered from Class 1E power sources. Accordingly, if the facility lost offsite power, an emergency power source would power the SDS valves and the pressurizer level indicators. This arrangement is accept-

able because it conforms with the intent of NUREG-0737. Use of SDS is further discussed in Section 6.7 and Chapter 19 of this report.

8.5 Station Blackout

The SBO rule (10 CFR 50.63) requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of a specified duration. Guidance for conformance to the SBO rule is in (1) RG 1.155, "Station Blackout," (2) the Nuclear Management and Resources Council, Inc. (NUMARC) 87-00 document, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," and (3) a supplement to NUMARC 87-00, "Supplemental Questions/Answers and Major Assumptions," dated December 27, 1989 (issued to the industry by NUMARC on January 4, 1990).

The NRC developed a policy issue in SECY-90-016, dated January 12, 1990, that was approved by the Commission on June 26, 1990, which requires that the evolutionary ALWRs meet the SBO rule by including an AAC power source (e.g., CTG) of diverse design capable of powering at least one complete set of normal shutdown loads and to back up the EDGs. EPRI has also included a requirement that a large-capacity, diverse AAC power source (e.g., CTG) with the capacity to power one complete set of normal safe-shutdown loads and to back up the EDG be part of the evolutionary ALWR design. Therefore, the staff's proposed applicable regulation as the standard design must provide an alternate ac power source for the purpose of dealing with station blackout.

ABB-CE committed to meet the SBO requirements by providing an AAC power source. ABB-CE stated that the AAC Source for System 80+ is a non-safety-grade combustion gas turbine provided to cope with a LOOP and an SBO scenario. This standby unit will meet the requirements in 10 CFR 50.63 by being independent and diverse from the Class 1E standby EDGs. The AAC source will not normally be directly connected to the plant's main or standby offsite power sources or to the Class 1E power distribution system, thus minimizing the possibility of a common-cause failure.

The CTG is designed to automatically start and be ready to accept load within two minutes from the onset of a LOOP event so that the plant will be capable of maintaining core cooling and containment integrity. The COL applicant will also store sufficient fuel on site to support 24 hours of CTG operation at rated load. A dedicated 125-V dc battery will power the instrumentation and controls necessary to start and run the AAC source.

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ABB-CE addressed periodic testing of the AAC power source and committed to require the COL applicant to establish an AAC QA program consistent with RG 1.155, Appendix A.

Therefore, a System 80+ plant will have a fully qualified CTG as an AAC power source. However, regarding core cooling for an SBO event, ABB-CE was required to confirm that,

- (1) The plant will have sufficient condensate storage to remove decay heat for the duration of an SBO in accordance with RG 1.155, Section 3.3.2.
- (2) The equipment and systems will be operable during an SBO event.
- (3) By confirming that the AAC source provides appropriate cooling, the areas in which SBO equipment is located will be habitable.

In the DSER, the staff stated it would verify that these three aspects of the design are included as design commitments. This was designated as DSER Open Item 8.5-1.

In Amendment Q, ABB-CE supplemented the information on SBO as follows:

ABB-CE indicated that the EFW system contains two, seismically designed, 100-percent-capacity emergency feedwater storage tanks (EFWSTs). Each tank has a safety-related condensate storage volume of 350,000 gallons, which is sufficient condensate to maintain the unit in hot standby condition for 8 hours by using the EFW turbine-driven pumps. The EFW system can either be manually or automatically actuated to operate during SBO conditions. The EFW system is configured into two separate divisions. Each division is aligned to feed its respective steam generator which consists of one EFWST, one 100-percent-capacity motor-driven pump and one 100-percent-capacity steam-driven pump. The EFW system can be controlled from either the MCR or the remote shutdown room. This addresses Item 1 (above) and is acceptable.

ABB-CE indicated that areas in which SBO equipment is located will be habitable, since the AAC source is fully rated to power the chillers, the required SBO area ventilation units, and other support/auxiliary equipment, and will be available for loading within 2 minutes of receiving its start signal. ABB-CE committed to NUMARC 87-00 to ensure that the environment during and following an SBO event will not exceed the environment for which the equipment is designed and qualified. This addresses Items 2 and 3 (above) and is acceptable.

The CTG for the System 80+ design has sufficient capacity and capability

to power either both sets of the permanent non-safety-related loads, or one set of permanent non-safety-related loads and one set of a safety division's loads. The CTG is designed to automatically start, accelerate to rated speed, reach nominal voltage, and to begin accepting loads within 2 minutes of receiving its start signal (i.e., following LOOP event). It is connected to the permanent non-safety buses by automatic connection, but has no automatic connection from the permanent non-safety buses to the safety buses. It is capable of being operated during and after an SBO without any ac support systems powered from the preferred power supply and is equipped with its own fuel system that is separate from the fuel supply for the onsite EDG. Its fuel will be sampled and analyzed according to applicable standards.

The AAC source is a self-contained unit equipped with its own auxiliary control and support systems. The AAC can be manually started, stopped, loaded, and controlled locally and from the MCR. The AAC is located within the protected area, outside the turbine missile impact zone; it is surrounded by a perimeter security fence and has an intrusion alarm system.

The CTG has a continuous rating capacity margin of at least 10 percent to compensate for load growth. The CTG will be tested at the factory, as is the Class 1E EDG, to demonstrate its ability to start, its ability to accelerate to rated speed and voltage, and its ability to supply power within 2 minutes. The CTG will be subject to site acceptance testing, periodic preventive maintenance, inspection, testing, and operational reliability assurance process goals (see Section 17.3 of this report for discussion of O-RAP). The AAC source will also be routinely inspected and maintained while the plant is at power.

On the basis of its review, the staff concludes that the CE System 80+ design for meeting the SBO rule through the use of a CTG (AAC power source) meets the guidelines of RG 1.155, and NUMARC 87-00, and satisfies the requirements of 10 CFR 50.63 the Commission policy on SBO (SECY-90-016), and the staff's proposed applicable regulation, and is, therefore, acceptable. On this basis, DSER Open Item 8.5-1 is resolved.

8.6 Combined License (COL) Action Items

ABB-CE committed to require periodic testing of electrical components in various electrical power systems. ABB-CE should have included in the CESSAR-DC those items that the COL applicant must address ("COL Action Items"). In the DSER, the staff stated that it will review this further

upon receiving the appropriate information. This was designated as DSER Open Item 8.6-1.

In Amendment Q, ABB-CE added a new CESSAR-DC Section 8.1.4.5, "COL Action Items," to address this issue. The staff has reviewed this information and concludes that all periodic testing of electrical components has been included in CESSAR-DC Section 8.1.4.5 as a COL action item. This satisfies the staff's concern and is acceptable. On this basis, DSER Open Item 8.6.1 is resolved.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New-Fuel Storage

The staff reviewed the new fuel storage capability in accordance with standard review plan (SRP) Section 9.1.1. Staff acceptance of the new fuel storage facility is based on the design complying with the requirements of General Design Criteria (GDC) 2 as it relates to the ability of structures housing the facility and the facility itself to withstand the effects of natural phenomena such as earthquakes; GDC 5 as it relates to shared structures, systems, and components (SSCs) important to safety being capable of performing required safety functions; GDC 61 as it relates to the facility design for fuel storage; and GDC 62 as it relates to the prevention of criticality. Compliance with GDC 2 is based on complying with the guidance of Regulatory Guide (RG) 1.29, Position C.1.L as it relates to the seismic classification of facility components. Specific criteria necessary to meet the requirements of GDC 61 and 62 are American Nuclear Society (ANS) 57.1, "Design Requirements for LWR Fuel Handling System," and ANS 57.3, "Design Requirements for New LWR Fuel Storage Facilities," as they relate to the prevention of criticality and to the aspects of the radiological design. In CESSAR-DC Section 9.1.1, ABB-CE presents the design bases, a description, and the safety evaluation of the new-fuel storage arrangement for the System 80+ design.

According to CESSAR-DC Section 9.1.1, the new fuel storage racks are made up of two 11 x 11, 50 percent-density, individual rack modules, each containing 121 cells. The storage racks are stainless steel and have rectangular fuel storage cells on a single square pitch of 24.84 cm (9.78 in.) for all cells. Cell blockers are installed in alternate cells of each storage rack to limit new fuel storage to a total of 121 fuel assemblies. The racks are bolted to embedments at the bottom of the rack storage cavity to preclude tipping.

In CESSAR-DC Section 9.1.1, ABB-CE provides a safety evaluation to show that the new fuel storage rack design complies with the design bases. The section contains an evaluation of criticality safety, which includes postulated accidents, criticality safety assumptions, and criticality safety margins. The postulated accidents considered are flooding, optimum moderation conditions, fuel-handling accidents, and excessive tensile loads on the rack. The design features used to minimize the possibility of these accidents are:

- hoist travel limits and crane interlocks

- limited load handling when installing or removing fuel from the racks
- racks designed for safe-shutdown earthquake (SSE) conditions
- racks designed for dropped fuel assembly (and handling tool) conditions
- new-fuel-handling crane designed to preclude the crane, or any part of the crane, from falling into the new-fuel-handling area

The staff reviewed CESSAR-DC Section 9.1.1 in conjunction with ABB-CE's responses to the staff's requests for additional information (RAI). The staff performed its review in accordance with the guidance and acceptance criteria in NUREG-0800, SRP Section 9.1.1. The staff directed its evaluation at determining whether or not the new-fuel storage design complies with the requirements of GDC 2, 5, 61, and 62. The staff reached the following conclusions:

- (1) Based on information provided in standard safety analysis report (SSAR) Section 9.1.1 and SSAR Table 3.2-1, the new fuel storage facility, including the storage cavity and racks, is designed and classified to seismic Category I requirements. Furthermore, the facility is housed in the Nuclear Annex building, a seismic Category I structure. Therefore, the essential portions of the new fuel storage facility are protected from the effects of earthquakes, floods, hurricanes, tornadoes, and other natural phenomena, as required by GDC 2. Based on this information, the staff concludes that the new fuel storage facility meets the guidelines of Position C.1.L of RG 1.29 and therefore complies with the requirements of GDC 2 regarding the ability of structures housing the facility and the facility itself to withstand the effects of natural phenomena such as earthquakes. The following related confirmatory issues were identified in the staff's draft safety evaluation report (DSER):
 - (a) In response to RAI Q410.103(b), ABB-CE stated that the seismic Category I facilities associated with new-fuel storage consist of the storage vault and the rack restraint system. The seismic category of other building components associated with handling fuel assemblies is noted in CESSAR-DC Table 3.2-1. The staff asked ABB-CE to incorporate this response into the CESSAR-DC. This was designated as DSER Confirmatory Item 9.1.1-1.

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Subsequently, ABB-CE incorporated the additional information in Amendment N to CESSAR-DC (Section 9.1.1.3.3). Seismic Category I components include the new-fuel storage racks, the storage vault, and the rack restraint system. The seismic classification of other components associated with handling fuel assemblies is given in CESSAR-DC Table 3.2-1. Components near the fuel storage area that are not seismic Category I are classified as seismic Category II to ensure that their failure will not damage the fuel or fuel racks. On this basis, DSER Confirmatory Item 9.1.1-1 is resolved.

- (b) In response to RAI Q410.103(i), ABB-CE discussed the capability of the system to withstand the failure of non-seismic Category I systems or structures located nearby. The staff asked ABB-CE to incorporate this response into the CESSAR-DC. This was designated as DSER Confirmatory Item 9.1.1-2.

Subsequently, ABB-CE incorporated the additional information in Amendment N to CESSAR-DC (Section 9.1.1.3.1.1.K) and in Table 3.2-1. As stated above, components near the fuel storage area that are not seismic Category I are classified as seismic Category II so that their failure will not damage the fuel or fuel racks. On this basis, DSER Confirmatory Item 9.1.1-2 is resolved.

- (c) In response to RAI Q410.103(c), ABB-CE presented additional descriptive information on fuel rack design features and on the seismic classification of equipment in the "new-fuel inspection area." The staff asked ABB-CE to incorporate this information into the CESSAR-DC, along with the function of the L inserts. This was designated as DSER Confirmatory Item 9.1.1-3.

Subsequently, ABB-CE incorporated this additional information in Amendment N to CESSAR-DC (Sections 9.1.1.2 and 9.1.1.3.1.1.K and Table 3.2-1). The seismic classification of fuel facilities has already been discussed. In addition, ABB-CE clarified that the L-inserts act as additional metal thickness for neutron absorption and limit the displacement between the fuel assemblies and the rack wall in order to minimize seismic acceleration loads. On this basis, DSER Confirmatory Item 9.1.1-3 is resolved.

- (2) Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the inde-

pendence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the combined license (COL) applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. This was designated as DSER COL Action Item 9.1.1-1. Upon further review, the staff has determined that the design described in CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the new fuel storage system complies with the requirements of GDC 5, and COL Action Item 9.1.1-1 is acceptable.

- (3) The staff asked ABB-CE to address the following open items before the staff completed its review of the design's compliance with the requirements of GDC 61 (regarding fuel storage, fuel handling, and radioactivity control) and GDC 62 (regarding criticality safety):

- (a) For the review of CESSAR-DC Section 9.1.1, the staff asked ABB-CE to submit criticality information, including numerical values of maximum k_{eff} for flooding, foam/mist immersion (including value of relative water density), and dropped fuel assembly accident scenarios including maximum height of dropped fuel. This was designated as DSER Open Item 9.1.1-1.

Subsequently, ABB-CE clarified that the K_{eff} for the new-fuel racks is maintained less than 0.95 should the area become flooded with pure, unborated water, and less than 0.98 should the racks become enveloped in an aqueous foam or mist of optimum density, assuming a maximum 5 percent fuel enrichment. The K_{eff} values include allowances for deviations from normal conditions and calculational uncertainties. Load drop analyses were performed to identify the worst-case load drop and ensure that the minimum spacing between fuel assemblies is maintained should this load drop occur. The worst-case load drop results in an impact energy of 10,519 newton-meters (N-m) (93,100 in.-lbs) to one edge of the a fuel rack cell. In this worst-case scenario, minimum fuel spacing was maintained. On this basis, DSER Open Item 9.1.1-1 is resolved.

- (b) In the DSER, the staff stated that SRP Section 9.1.1, Item III 2.a, requires criticality information to show the racks in a subcritical condition, that is, k_{eff} of less than about 0.95, with design enrichment fuel. This requirement is in addition to the requirement for k_{eff} to be less than 0.95 under fully flooded conditions and less than 0.98 under optimum moderation (form/mist) conditions. This was designated as DSER Open Item 9.1.1-2.

As stated above, ABB-CE clarified that a fuel enrichment value of 5 percent was assumed in the criticality analysis (see CESSAR-DC Section 9.1.1.3.1.3) to form an upper bound which would encompass any fuel management scheme. On this basis, DSER Open Item 9.1.1-2 is resolved.

- (c) In the DSER, the staff stated that ABB-CE should discuss which accidents require the use of computer code DOT-4 and which require the use of the KENO IV code, and should also discuss the respective accuracies of these codes. The staff stated that this discussion should be broader than that provided in Amendment E of CESSAR-DC, which merely indicated that KENO IV is used for geometries involving three-dimensional input. This was designated as DSER Open Item 9.1.1-3.

Subsequently, ABB-CE clarified that the KENO IV code is used for criticality analyses for new fuel during accident conditions (both flood and mist); the DOT-4 code is used to perform criticality analyses for spent fuel for normal conditions; and KENO IV is used to determine k_{eff} for the spent fuel during accident conditions. The criticality calculations allow for deviations from normal conditions (i.e., variations in water temperature) and for calculational uncertainties. On the basis of this additional information, DSER Open Item 9.1.1-3 is resolved.

- (d) In the DSER, the staff stated that ABB-CE should discuss the assumption inherent in the statement that a 25.4 cm (10 in.) space between the top of the active fuel and the top of the fuel rack precludes any criticality in the event of a fuel assembly drop. In addition, the staff stated that the discussion should indicate the fuel enrichment considered. This was designated as DSER Open Item 9.1.1-4.

Subsequently, ABB-CE submitted additional information clarifying that the load drop analysis ensured that the minimum spacing between fuel assemblies would be maintained given a worst-

case load drop. This analysis included maintaining the 25.4 cm (10 in.) spacing between the top of the active fuel and the top of the fuel rack when a fuel assembly is dropped. On the basis of this additional information, DSER Open Item 9.1.1-4 is resolved.

- (e) One of the postulated accidents presented in CESSAR-DC Section 9.1.1.3.1.1 is flooding of the new-fuel storage vault by unborated, room-temperature water. In CESSAR-DC Section 9.1.1.3.1.3, ABB-CE states that spatial calculations were performed for a selection of uniform water densities covering the ranges in which reactivity peaks occur. Because flooding is considered to be an external natural phenomenon, room-temperature water cannot be assumed. Therefore, in the DSER, the staff stated that ABB-CE should confirm that flooding with room-temperature water is conservative for an optimum moderation condition. This was designated as DSER Open Item 9.1.1-5.

Subsequently, ABB-CE submitted Amendment Q to CESSAR-DC which clarified that the criticality analysis for the new fuel considered a range of water densities (including the worst-case water density) in addition to flood and mist of unborated water at room temperature. The analysis concludes that k_{eff} is maintained below 0.95 during flooded conditions and below 0.98 during foam and mist conditions. On the basis of this clarifying information, DSER Open Item 9.1.1-5 is resolved.

- (f) As required by SRP Section 9.1.1, Item III-2-b, the staff stated in the DSER that ABB-CE should discuss provisions in the design for draining the vault to prevent the accumulation of a fluid moderator. This was designated as DSER Open Item 9.1.1-6.

Subsequently, ABB-CE submitted Amendment N to the CESSAR-DC which clarified that a drain line at the bottom of the storage vault directs fluid to the floor drain sump. In addition, the drain line contains a non-return check valve to prevent fluid backflow into the vault. On the basis of this additional information, DSER Open Item 9.1.1-6 is resolved.

- (g) In the DSER, the staff stated that ABB-CE should perform an analysis to ensure that the failure of non-seismic Category I systems or structures located in the vicinity of the new-fuel storage

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racks cannot cause an increase in k_{eff} beyond the maximum allowable k_{eff} . The staff stated that ABB-CE should either submit such an analysis to the staff or require the COL applicant to provide the analysis at the COL license stage. This was designated as DSER Open Item 9.1.1-7.

As discussed previously, ABB-CE clarified that components near the fuel storage area that are not seismic Category I are classified as seismic Category II so that their failure will not damage the fuel or fuel racks. On the basis of this information, DSER Open Item 9.1.1-7 is resolved.

- (h) In CESSAR-DC Section 9.1.1.3.1.2.D ABB-CE states that the array is assumed to be surrounded on each face by a 2-foot-thick, close-fitting reflector of concrete. The walls, floor, and material above the fuel are assumed to be several inches away from the array. In the DSER, the staff stated that ABB-CE must show how these distances between concrete and fuel array produce the most conservative assumptions for the criticality analysis. This was designated as DSER Open Item 9.1.1-8.

Subsequently, ABB-CE submitted Amendment N to CESSAR-DC which clarified that the criticality analyses assumed a close-fitting concrete reflector on all six sides of the racks. This is conservative since the walls surrounding the racks are actually several inches away from the racks. On the basis of this additional information, DSER Open Item 9.1.1-8 is resolved.

- (i) In the DSER, the staff stated that ABB-CE must explain how cross-placement of new-fuel and spent-fuel storage racks will be prevented. This was designated as DSER Open Item 9.1.1-9.

Subsequently, the staff reviewed Amendment Q to CESSAR-DC. In CESSAR-DC Section 9.1.4, ABB-CE discusses operational and design aspects of the fuel-handling equipment. During refueling, new fuel can be placed in region I of the spent fuel racks in preparation for transfer of the new-fuel into the reactor building. Spent fuel, however, cannot inadvertently be transferred to the new-fuel storage area. Before refueling operations begin, the new-fuel is removed from the new-fuel storage racks and transferred to the new-fuel elevator by using the fuel building handling crane and the short fuel handling tool. The new-fuel elevator then lowers the new-fuel assembly into the spent fuel pool to allow the spent fuel

handling machine to transfer the fuel assembly to the region I area of the spent fuel racks or the transfer system upender. A hoist lockout interlock on the new-fuel elevator prevents raising the elevator with a fuel assembly in the elevator box. This interlock is a backup for the administrative control which prohibits the placement of a spent-fuel assembly in the new-fuel elevator. On the basis of this information, DSER Open Item 9.1.1-9 is resolved.

In addition to the open items listed above, the staff also identified the following general confirmatory issues in the DSER:

- (1) In response to RAI Q410.103(a), ABB-CE stated that the applicable portions of RG 1.13, "Spent Fuel Storage Facility Design Basis," have been complied with, and are defined in CESSAR-DC Sections 9.1.1.1.A and 9.1.1.1.C. In the DSER, the staff stated that the criticality design basis described in Section 9.1.1.1.A is not derived from RG 1.13. The staff stated that ABB-CE must revise CESSAR-DC Section 9.1.1.1.B to specify which parts of RG 1.13 are complied with. This was designated as DSER Confirmatory Item 9.1.1-4.

Subsequently, ABB-CE incorporated this information in Amendment N to CESSAR-DC. The design conforms to all the guidelines of RG 1.13, except those regarding the spent-fuel pool water supply. These guidelines are not applicable to the new-fuel storage facilities since the fuel is stored in a dry condition. On the basis of this information, Confirmatory Item 9.1.1-4 is resolved.

- (2) In response to RAI Q410.103(d), ABB-CE gave the design basis for the storage capacity of 121 new-fuel assemblies. In the DSER, the staff stated that ABB-CE must incorporate this information into CESSAR-DC Section 9.1.1.1.D. This was designated as Confirmatory Item 9.1.1-5 in the DSER.

Subsequently, ABB-CE incorporated this information in Amendment N to CESSAR-DC. The capacity represents 50 percent of the fuel assemblies in the core and envelopes any reload batch size that would occur for refueling cycle lengths up to 24 months. On the basis of this information, DSER Confirmatory Item 9.1.1-5 is resolved.

- (3) In response to RAI Q410.103(e), ABB-CE stated that the design is in compliance with ANS 57.1, "Design Requirements for Light Water Reactor

Fuel Handling Systems," and ANS 57.3, "Design Requirements for New Fuel Storage Facilities." In the DSER, the staff stated that ABB-CE must incorporate this statement into CESSAR-DC Section 9.1.1.1. This was designated as DSER Confirmatory Item 9.1.1-6.

Subsequently, ABB-CE submitted Amendment N to the CESSAR-DC which stated that the design of the fuel-handling equipment complies with the guidelines of ANS 57.1 and ANS 57.3. The new-fuel storage racks, storage vault, and rack restraint system are designed to seismic Category I requirements. Equipment in the vicinity of the new-fuel storage area that is not designed to seismic Category I standards is designed so that its failure will not damage the fuel racks or the fuel. Furthermore, load-handling equipment associated with the new-fuel storage facilities is designed to hold its load during a seismic event or loss of power. On the basis of this additional information, DSER Confirmatory Item 9.1.1-6 is resolved.

- (4) In response to RAI Q410.103(f), ABB-CE gave the restrictions limiting the lifting capacity of the overhead crane. In the DSER, the staff stated that ABB-CE must incorporate this information into CESSAR-DC Section 9.1.1.3.1, along with a statement that excessive uplifting force cannot be applied. This was designated as DSER Confirmatory Item 9.1.1-7.

Subsequently ABB-CE submitted Amendment N to the CESSAR-DC which clarified that the new-fuel-handling hoist on the fuel building overhead crane is limited in its lifting capacity by either adjusting the motor stall torque or by using load-limiting devices. The staff concludes that these provisions will ensure that excessive loads will not be imposed on the assemblies. On the basis of this additional information, DSER Confirmatory Item 9.1.1-7 is resolved.

- (5) In response to RAI Q410.103(g), ABB-CE discussed the design features that will prevent heavy objects from falling onto the new fuel racks. In the DSER, the staff stated that ABB-CE must incorporate the information into CESSAR-DC Section 9.1.1.3.1.1. This was designated as DSER Confirmatory Item 9.1.1-8.

Subsequently, ABB-CE submitted Amendment N to the CESSAR-DC which incorporated the required information. The new-fuel racks and the spent-fuel pool are located at the opposite ends of the fuel building to eliminate the possibility of moving heavy loads (e.g., a spent-fuel shipping cask) over the new-fuel assem-

blies. Also, any components whose failure could damage fuel is classified as seismic Category II to preclude their failure and entry into the new-fuel storage area. Finally, no load can be moved over the fuel whose impact energy, if dropped from its operating elevation, exceeds that of the worst-case load considered in the load-drop analyses. On the basis of this information, DSER Confirmatory Item 9.1.1-8 is resolved.

- (6) In the DSER, the staff stated that ABB-CE must clarify in CESSAR-DC Section 9.1.1.1 what fraction of the total core is represented by the 121 new-fuel assemblies. This was designated as DSER Confirmatory Item 9.1.1-9.

Subsequently, ABB-CE submitted Amendment N to the CESSAR-DC which clarified that 121 new-fuel assemblies represent 50 percent of the fuel assemblies in the core. On the basis of this additional information, DSER Confirmatory Item 9.1.1-9 is resolved.

- (7) In the DSER, the staff stated that ABB-CE must clarify the design requirements for the new-fuel racks for SSE conditions and dropped fuel assembly conditions. This was designated as DSER Confirmatory Item 9.1.1-10.

As was stated earlier, ABB-CE submitted Amendment Q to the CESSAR-DC which clarified that the seismic Category I components consist of the new-fuel storage racks, the storage vault, and the rack restraint system. The seismic classification of other components associated with handling fuel assemblies is given in CESSAR-DC Table 3.2-1. Components near the fuel storage area that are not seismic Category I are classified as seismic Category II so that their failure will not damage the fuel or fuel racks. On the basis of this information, DSER Confirmatory Item 9.1.1-10 is resolved.

On the basis of the information submitted by ABB-CE, the staff concludes that the design meets the guidelines of ANS 57.1 and ANS 57.3 and the requirements of GDC 61 and 62 as they relate to fuel storage and handling and radioactivity control.

The new-fuel storage facility includes the fuel assembly storage racks and the concrete storage vault that contains the storage racks. On the basis of its review of ABB-CE's proposed design criteria, design bases, and safety classification for the new-fuel storage facility regarding the provisions necessary to maintain a subcritical array, the staff concludes that the design of the new-fuel storage facility is acceptable and complies with the requirements of GDC 2, 5, 61, and 62.

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9.1.2 Spent-Fuel Storage

The staff reviewed the spent fuel storage capability in accordance with SRP Section 9.1.2. Staff acceptance of the spent fuel storage facility is based on the design complying with the requirements of GDC 2 as it relates to the ability of structures housing the facility and the facility itself to withstand effects of natural phenomena such as earthquakes, tornadoes, and hurricanes; GDC 4 as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of external missiles, and internally-generated missiles, pipe whip, jet impingement forces, and adverse environmental conditions associated with pipe breaks, such that safety functions will not be impaired; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; GDC 61 as it relates to the facility design for fuel storage and handling of radioactive materials; GDC 62 as it relates to the prevention of criticality; and GDC 63 as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions.

Compliance with the requirements of GDC 2 is based on complying with the guidance of RG 1.13, Position C.3, the applicable portions of RG 1.29, RG 1.117, and ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," paragraphs 5.1.1, 5.1.3, 5.1.12, 5.3.2, and 5.3.4. Compliance with the requirements of GDC 4 is based on complying with the guidance RG 1.13, Position C.3, RGs 1.115 and 1.117, and the appropriate paragraphs of ANS 57.2. Compliance with the requirements of GDC 61 is based on complying with the guidance of RG 1.13, Positions C.1 and C.4 and the appropriate paragraphs of ANS 57.2 as well as meeting the fuel storage capacity guidelines noted in subsection III.1 of SRP 9.1.2. Compliance with the requirements of GDC 62 is based on complying with the guidance of RG 1.13 Positions C.1 and C.4 and the appropriate paragraphs of ANS 57.2. Finally, compliance with the requirements of GDC 62 is based on complying with the guidance of paragraph 5.4 of ANS 57.2

In CESSAR-DC Section 9.1.2, ABB-CE presents design bases, a description, and the safety evaluation of the spent-fuel storage arrangement.

According to CESSAR-DC Section 9.1.2, the spent-fuel storage racks are made up of twelve 11 x 11 individual stainless steel modules, each containing 121 cells. Cell blockers are installed to provide 50-percent density storage in region I (for 363 spent fuel assemblies) and 75-percent density storage in region II (for 544 spent fuel assemblies). L-inserts made of non-poisoned stainless steel are used to

provide a flux trap water gap in the usable cells. Neutron-poisoned inserts may be used for future expansion potential. The spent-fuel storage racks are installed in the spent-fuel pool, which is a stainless steel-lined, concrete-walled pool that is an integral part of the fuel building.

In CESSAR-DC Section 9.1.2.3, ABB-CE presents a safety evaluation to show that the spent-fuel storage rack design and location comply with its design bases. The safety evaluation includes postulated accidents, criticality safety assumptions, and criticality safety margins. The postulated accidents considered are fuel-handling accidents, excessive tensile loads on the rack, and a misplaced fuel assembly. The design features used to minimize the possibility of these accidents are:

- hoist travel limits and crane interlocks
- limited load handling when installing or removing fuel from the racks
- racks designed for SSE conditions
- racks designed for dropped fuel assembly (and handling tool) conditions
- fuel-handling machine designed to seismic Category II requirements to preclude the machine, or any part of the machine, from falling into the spent-fuel pool

The staff reviewed CESSAR-DC Section 9.1.2 in conjunction with ABB-CE's responses to the staff's RAIs. The staff based its review on guidance and acceptance criteria in SRP Section 9.1.2. The staff directed its evaluation at determining whether or not the spent fuel storage facility complies with the requirements of GDC 2, 4, 5, 61, 62, and 63. Acceptability for meeting these criteria is based on conformance to Positions C.1, C.3, and C.4 of RG 1.13; applicable portions of RG 1.29, "Seismic Design Classification"; RG 1.117, "Tornado Design Classification"; and the appropriate paragraphs of ANS 57.2, "Design Objectives for Light Water Spent Fuel Storage Facility at Nuclear Power Plants." The staff concludes the following:

- (1) The design complies with the requirements of GDC 2 by conforming to Position C.3 of RG 1.13, the applicable portions of RGs 1.29 and 1.117, and applicable paragraphs of ANS 57.2.

Heavy loads are prevented from being lifted over the spent-fuel pool by electrical and mechanical interlocks. In addition, the fuel racks are designed to withstand a load drop equivalent to that from a fuel assembly and

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its associated handling tool when dropped from its operating height.

The spent-fuel storage racks are housed in the fuel-handling portion of the nuclear annex, a seismic Category I structure designed to protect systems and equipment from external floods, missiles, earthquakes, and tornados. The spent-fuel pool and racks are designed to seismic Category I requirements. Components which are not designed to seismic Category I standards and whose failure could damage the fuel or safety-related systems and equipment are designed to ensure that they will not fail during a seismic event, are seismically restrained, or are removed from the area during normal operation (see CESSAR-DC Section 9.1.4.2.3.3).

The spent-fuel storage area is designed to prevent a loss of pool water from uncovering the spent fuel. This is accomplished by locating pipe penetrations at least 10 feet above the top of the fuel assemblies. The bottom of the gates that separate the spent-fuel pool from the spent fuel cask laydown area and from the fuel-transfer canal are located above the top of the fuel assemblies and are oriented so that water pressure in the spent-fuel pool area will keep the gates closed when the cask laydown area or transfer canal is drained. The racks are designed to withstand the drop of a fuel assembly and its associated handling tool, and heavy loads are prevented from traversing over the spent-fuel pool.

On the basis of this information, the staff concludes that the System 80+ design complies with the guidelines of Position C.3 of RG 1.13; Positions C.1 and C.2 of RG 1.29; RG 1.117; and ANS 57.2. The design, therefore, complies with the requirements of GDC 2 regarding protection of the spent fuel storage facilities from the effects of natural phenomena.

- (2) The design complies with the requirements of GDC 4 regarding the environmental and missile protection design basis by conforming to Position C.3 of RG 1.13, the applicable portions of RGs 1.115, "Protection Against Low-Trajectory Turbine Missiles," and 1.117; and appropriate paragraphs of ANS 57.2.

The protective interlocks already discussed protect the spent-fuel storage area from gravitational missiles. In addition, the fuel will be protected from turbine missiles by proper orientation of the turbine, as well as by features to prevent the generation of turbine missiles. Turbine missiles are discussed in Section 3.5.1.3 of the CESSAR-DC and are evaluated in Section 3.5.1.3 of this report.

The location (seismic Category I - designed nuclear annex) and design (seismic Category I) of the spent-fuel storage facility ensures that the spent fuel will not be damaged by externally generated missiles or by missiles generated by natural phenomena. Redundant divisions of safety-related equipment are separated by a divisional wall which ensures that an internally generated missile in one division will not damage equipment in the other division. These missiles are discussed in Sections 3.5.1 and 3.5.2 of the CESSAR-DC and are evaluated in Sections 3.5.1 and 3.5.2 of this report.

The spent fuel storage racks are protected from the effects of pipe failures as discussed in Sections 3.6.1 and 3.6.2 of the CESSAR-DC. Protection of SSCs from the effects of pipe failures is evaluated in Sections 3.6.1 and 3.6.2 of this report. In the staff's DSER, the following open item was identified:

In response to RAI Q410.104a, ABB-CE explained how the design complies with the guidance given in SRP Section 9.1.2. In the DSER, the staff stated that ABB-CE must incorporate the response into CESSAR-DC Section 9.1.2, expanding it to show how the design conforms to Position C.3 of RG 1.13, relative to crane interlocks. This was designated as DSER Open Item 9.1.2-1.

Subsequently, ABB-CE submitted Amendment Q to the CESSAR-DC which clarified that the fuel-handling equipment will have interlocks to prevent hoisting spent fuel above a maximum lift height and to ensure that the spent-fuel cask cannot be moved over the spent-fuel. On the basis of this additional information, DSER Open Item 9.1.2-1 is resolved.

On the basis of the above information, the staff concludes that the ABB-CE System 80+ design complies with the guidelines of Position C.3 of RG 1.13, RGs 1.115 and 1.117, and ANS 57.2 and, therefore, complies with the requirements of GDC 4 regarding protection of the spent fuel from the effects of missiles and pipe failures.

- (3) Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the combined license (COL) applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Struc-

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tures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. This was designated as DSER COL Action Item 9.1.2-1. Upon further review, the staff has determined that the design described in CESSAR-DC does not share SSCs with other nuclear power units. Therefore, the spent-fuel-storage facility complies with the requirements of GDC 5, and DSER COL Action Item 9.1.2-1 is resolved.

- (4) In the DSER, the staff stated that the design was consistent with the requirements of GDC 61 and 62 regarding spent-fuel storage, handling, criticality, and radioactivity control.

The total spent-fuel storage capacity represents 376 percent of a full core. The facilities are designed to seismic Category I requirements and are housed in the spent-fuel area of the nuclear annex. This portion of the nuclear annex is served by the fuel building ventilation system (FBVS) (see Section 9.4 of the CESSAR-DC). As identified in CESSAR-DC Table 3.2-1, the FBVS is seismic Category I, Safety Class 3, and is designed to accommodate a postulated fuel-handling accident. The load-handling equipment is designed to ensure that fuel cannot be raised to a height that would result in a loss of the water shield for the fuel. Radiation monitors are arranged to detect excessive radiation levels in the spent fuel pool area and uninterrupted communications are available.

The staff identified the following open items that needed to be addressed:

- (a) ABB-CE should discuss which analyses require the use of computer code DOT-4 and which require the use of the KENO IV code, and should also discuss the respective accuracies of these codes. This discussion should be broader than the discussion in Amendment E, which merely indicates that KENO IV is used for geometries involving three-dimensional input. This was designated as DSER Open Item 9.1.2-2.

Subsequently, ABB-CE clarified that the DOT-4 code is used to calculate K_{eff} in the spent-fuel storage racks for both regions I and II for normal design conditions. The KENO IV code is used to calculate K_{eff} for a postulated dropped fuel assembly. With this additional information, DSER Open Item 9.1.2-2 is resolved.

- (b) ABB-CE should submit numerical values for k_{eff} for all normal design and accident analyses per-

formed for the spent-fuel storage facility. This was designated as DSER Open Item 9.1.2-3.

Subsequently, ABB-CE clarified that the K_{eff} for the spent-fuel racks is maintained less than 0.95 during both normal and accident conditions. The criticality analysis for region I of the spent-fuel racks during normal conditions takes no credit for boron, assumes a 5-percent fuel enrichment, and allows for uncertainties due to deviations from nominal conditions (e.g., variations in fuel rack pitch, rack thickness, and water temperature), as well as calculational uncertainties. For criticality calculations for region II of the spent-fuel racks during normal conditions, K_{eff} is calculated for various combinations of fuel enrichment and fuel burnups with allowances for deviations from nominal conditions and calculational uncertainties. For the dropped-fuel-assembly accident, the dropped-fuel assembly is assumed to contain 5-percent enriched fuel with a boron concentration that is 50 percent of the minimum concentration required by technical specifications (TSs). On the basis of this information, DSER Open Item 9.1.2-3 is resolved.

- (c) ABB-CE should submit the results of criticality analyses, which assume a boron concentration of 0 ppm for the dropped load conditions discussed in CESSAR-DC Sections 9.1.2.3.1.2.C, 9.1.2.3.1.3.A, and 9.1.2.3.1.3.B. This was designated as DSER Open Item 9.1.2-4.

As has been previously discussed, a 0-ppm boron concentration was assumed for the criticality analyses for normal conditions, and a boron concentration 50 percent of that required by TSs is assumed during accident conditions. The staff concludes that such concentrations under these conditions are sufficiently conservative for the criticality analyses. On the basis of this information, DSER Open Item 9.1.2-4 is resolved.

On the basis of the additional information submitted, the staff concludes that the design of the spent-fuel storage facility complies with the requirements of GDC 61 and 62 regarding spent fuel storage, handling, criticality, and radioactivity control.

- (5) The design complies with the requirements of GDC 63 regarding the monitoring of the status of the stored spent fuel by conforming to ANS 57.2. The following related confirmatory item was identified in the DSER:

In response to RAI O410.104d, ABB-CE presented the design features that comply with Paragraph 5.4 of ANS 57.2. ABB-CE should incorporate this information into CESSAR-DC Section 9.1.2.3. This was designated as DSER Confirmatory Item 9.1.2-1.

Subsequently, ABB-CE submitted Amendment Q to CESSAR-DC which contained information showing that the design contains adequate instrumentation to monitor pool water level and temperature, area radiation levels, ventilation sampling, and uninterruptible communications. On the basis of this additional information, DSER Confirmatory Item 9.1.2-1 is resolved.

On the basis of this information, the staff concludes that the spent-fuel storage facilities comply with the guidelines of ANS 57.2 and, therefore, comply with the requirements of GDC 63 regarding monitoring of the spent fuel.

The staff identified the following additional open items in the DSER:

- (1) ABB-CE must submit information which demonstrates how the spent-fuel storage pool will be protected from a spent-fuel storage cask drop. This was designated as DSER Open Item 9.1.2-5.

Subsequently, ABB-CE submitted clarifying information in CESSAR-DC Amendments I, L, and N. The spent-fuel storage cask is lifted and transported by the cask handling hoist on the fuel building overhead crane. Mechanical stops are installed on the bridge rails to prohibit the hoist from traveling over the spent-fuel pool area. In addition, limit switches, electrical and mechanical interlocks, and operating procedures (to be supplied by the COL applicant) are used to prevent an inadvertent cask drop which could damage spent-fuel. Finally, the spent-fuel cask laydown area is separated from the spent-fuel pool by a recessed gate and a reinforced-concrete wall. The gate is closed, locked, and sealed during all cask-handling operations. The floor in the laydown area is designed to withstand a cask drop from 9.14 m (30 ft). The pool gates are designed to use the water pressure in the pool to aid in closure. On the basis of this information, the staff concludes that the design of the pool protects the fuel assemblies from a storage cask drop. On this basis, DSER Open Item 9.1.2-5 is resolved.

- (2) ABB-CE must explain how leakage from the pool will be detected, and must discuss design features to prevent water inventory loss beyond minimum safe shielding and cooling limits. This was designated as DSER Open Item 9.1.2-6.

ABB-CE subsequently submitted the required information in CESSAR-DC Amendment L. All connections to the pool terminate at a level sufficient to ensure that, should the line break, the pool will not be drained below the minimum safe shielding level. In addition, pool connections contain anti-siphon holes to prevent siphoning of the pool water and isolation valves. Also, the pool has multiple sources of makeup water. Finally, leakage from the spent-fuel pool is detected by changes in pool level and area radiation monitors. Control room alarms actuate on indication of low pool level and high radiation. On the basis of this information, the staff concludes that the design of the spent-fuel pool provides assurance that leakage from the pool will be detected and sufficient water will be available to keep the assemblies cool and protect personnel from excessive radiation. On this basis, DSER Open Item 9.1.2-6 is resolved.

- (3) ABB-CE must explain how fuel racks will be anchored and prevented from tipping in the pool. This was designated as DSER Open Item 9.1.2-7.

Subsequently, ABB-CE verified that the rack design is similar to the System 80 design used at Palo Verde, which allows movement during SSEs. The staff has previously approved this design. On this basis, DSER Open Item 9.1.2-7 is resolved.

- (4) ABB-CE must explain how placement of a fuel assembly in an incorrect location is prevented. This was designated as DSER Open Item 9.1.2-8.

Subsequently, ABB-CE clarified in CESSAR-DC Amendment Q that inadvertent placement of a fuel assembly in an incorrect location is prevented by cell blockers in the racks, electrical and mechanical interlocks on fuel handling equipment, and administrative procedures. On the basis of this additional information, DSER Open Item 9.1.2-8 is resolved.

In addition, the staff identified the following general confirmatory issues in the DSER:

- (1) In response to RAI Q410.104b, ABB-CE listed the facilities that are mentioned in CESSAR-DC Section 9.1.2.3.2. ABB-CE would incorporate this list into CESSAR-DC Section 9.1.2.3.3. This was designated as DSER Confirmatory Item 9.1.2-2.

ABB-CE subsequently submitted this information in CESSAR-DC Amendment Q. The spent-fuel pool storage facility includes the spent-fuel storage racks, the spent-fuel pool concrete structure, the spent-fuel

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rack support system, and the pool liner. On the basis of this additional information, DSER Confirmatory Item 9.1.2-2 is resolved.

- (2) In response to RAI Q410.104c, ABB-CE presented justification that excessive force cannot be applied to the spent-fuel racks and fuel pool. ABB-CE should incorporate this justification into CESSAR-DC Section 9.1.2. This was designated as DSER Confirmatory Item 9.1.2-3.

This information was subsequently submitted in CESSAR-DC Amendment E. The spent-fuel handling machine contains a hoist overload interlock which interrupts hoisting of a spent-fuel assembly if the load increases above the overload setpoint. In addition, the hoist motor stall torque is limited so that the load on the cable is less than the allowable fuel assembly tensile load. On the basis of this additional information, the staff concludes that the fuel handling equipment cannot apply excessive loads to the fuel racks and fuel pool. On this basis, DSER Confirmatory Item 9.1.2-3 is resolved.

- (3) Portions of CESSAR-DC Section 9.1.2.3.1.3 are missing. ABB-CE should submit the missing pages. This was designated as DSER Confirmatory Item 9.1.2-4. The missing information was added in CESSAR-DC Amendment J. On this basis, DSER Confirmatory Item 9.1.2-4 is resolved.
- (4) In response to RAI Q410.64b, ABB-CE submitted information on the design criteria and anticipated controls to be used for storing fuel in region II. ABB-CE committed to insert material in CESSAR-DC Section 9.1.2.2.2 concerning the restrictions of fuel storage in region II to be imposed by administrative controls developed and implemented by the COL applicant. ABB-CE should revise CESSAR-DC Section 9.1.2.2.2 as indicated. This was designated as DSER Confirmatory Item 9.1.2-5.

ABB-CE subsequently submitted this information in CESSAR-DC Amendment N. Both region I and II are designed to accommodate fuel assemblies with initial fuel enrichment of up to 5 weight percent U-235. While region I has no restriction on burnup history of stored fuel assemblies, region II is restricted to storage of fuel assemblies having a minimum cumulative burnup which is dependent upon the initial enrichment of each fuel assembly. This restriction is to be developed and implemented by the COL applicant. On the basis of the incorporation of this information in the CESSAR-DC, DSER Confirmatory Item 9.1.2-5 is resolved.

- (5) In response to RAI Q410.54, ABB-CE showed compliance with SRP Section 9.1.2, Issue III.2.e, guidance. The staff asked ABB-CE to incorporate this material into CESSAR-DC Sections 9.1.2 and 9.1.4. This was designated as DSER Confirmatory Item 9.1.2-6.

Subsequently, ABB-CE incorporated the required information in CESSAR-DC Amendment Q. Spent-fuel is protected from the effects of a load drop by prohibiting the movement of any load over the fuel whose impact energy if dropped from its operating height exceeds that of the worst-case load drop postulated in the load drop analysis. This is accomplished by electrical and mechanical interlocks on fuel-handling equipment, as well as by using proper fuel-handling procedures. On the basis of the incorporation of this information in the CESSAR-DC, DSER Confirmatory Item 9.1.2-6 is resolved.

- (6) In response to RAI Q410.64a, ABB-CE submitted the required discussion on the failure of non-safety-related systems, and structures and their potential effects on the integrity and coolability of the spent-fuel racks. The staff asked ABB-CE to incorporate this response into CESSAR-DC Section 9.1.2. This was designated as DSER Confirmatory Item 9.1.2-7. ABB-CE subsequently incorporated this information in CESSAR-DC Amendment Q. Most equipment in the spent-fuel pool area is classified as seismic Category I or II, thus, providing protection for the spent-fuel from the effects of equipment failure during a seismic event. In addition, all loose components which are not seismically qualified (e.g., tools) are either removed from the reactor building and fuel area of the nuclear annex or are seismically restrained during reactor operation. Fuel-handling procedures require that ladders be used for loose components that are used for a particular short-term application, such as repair work. All permanently installed components are secured with locking devices or restraints to prevent them from becoming loose and falling into the refueling pool or the spent-fuel pool. On the basis of this information, the staff concludes that non-safety-related components will not pose a threat to the spent-fuel assemblies; therefore, DSER Confirmatory Item 9.1.2-7 is resolved.
- (7) ABB-CE submitted clarifying information regarding the two spent-fuel regions, in response to RAI Q410.64. This response is sufficient to adequately describe these regions. Incorporation of this information into the CESSAR-DC text was designated as DSER Confirmatory Item 9.1.2-8.

ABB-CE subsequently incorporated this information in CESSAR-DC Amendment N. Region I provides core offload capability and 50-percent density storage for approximately 363 spent-fuel assemblies (150 percent of a full core). Region II provides 75-percent density storage for approximately 544 spent-fuel assemblies (226 percent of a full core). Region I is also used for storing new-fuel assemblies in preparation for refueling outages and for storing defective fuel assemblies. In CESSAR-DC Section 9.1.2.2.2, ABB-CE provides a fuller discussion of the spent-fuel pool storage racks. On the basis of the incorporation of this information into the CESSAR-DC, DSER Confirmatory Item 9.1.2-8 is resolved.

- (8) ABB-CE must explain what fraction of a full core is represented by region II and the total spent-fuel storage capacity. This was designated as DSER Confirmatory Item 9.1.2-9. As stated above, ABB-CE submitted this information in CESSAR-DC Amendment N, Section 9.1.2.2.2. On the basis of the incorporation of this information, DSER Confirmatory Item 9.1.2-9 is resolved.
- (9) In the response to RAI Q410.66, ABB-CE submitted clarifying information regarding the protection of spent-fuel from a spent-fuel cask load drop. Incorporation of this information into the CESSAR-DC text was designated as DSER Confirmatory Item 9.1.2-10.

As discussed above as part of the resolution to DSER Open Item 9.1.2-5, ABB-CE submitted clarifying information in CESSAR-DC Amendments I, L, and N. The spent-fuel storage cask is lifted and transported by the cask handling hoist on the fuel building overhead crane. Mechanical stops are installed on the bridge rails to prevent the hoist from traveling over the spent-fuel pool area. In addition, limit switches, electrical and mechanical interlocks, and operating procedures (to be supplied by the COL applicant) are used to prevent an inadvertent cask drop which could damage spent-fuel. Finally, the spent-fuel cask laydown area is separated from the spent-fuel pool by a recessed gate and a reinforced concrete wall. The gate is closed, locked, and sealed during all cask handling operations. The floor in the laydown area is designed to withstand a cask drop from 9.14 m (30 ft). The pool gates are designed to use the water pressure in the pool to aid in closure. On the basis of this information, the staff concludes that the design of the pool protects the fuel assemblies from a storage cask drop.

The spent-fuel storage facility includes the spent-fuel storage racks, the spent-fuel storage pool that contains the

racks, and the associated equipment storage pits. On the basis of its review of ABB-CE's proposed design criteria, design bases, and safety classification for the spent-fuel storage facility and the provisions necessary to maintain a subcritical array, the staff concludes that the spent-fuel storage design complies with the guidelines of Section SRP 9.1.2 and the requirements of (1) GDC 2 as it relates to structures housing the facility, and the facility itself being capable of withstanding the effects of natural phenomena; (2) GDC 4 as it relates to structures housing the facility, and the facility itself, being capable of withstanding the effects of environmental conditions, missiles, pipe whip, and jet impingement forces associated with pipe breaks; (3) GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; (4) GDC 61 as it relates to the facility design for fuel storage and handling of radioactive materials; (5) GDC 62 as it relates to prevention of inadvertent criticality; and (6) GDC 63 as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions. The design, therefore, is acceptable.

9.1.3 Spent-Fuel Pool Cooling and Pool Purification System

The staff reviewed the spent fuel pool cooling and purification system (PCPS) in accordance with SRP Section 9.1.3. Staff acceptance of the PCPS design is based on the design complying with the requirements of GDC 2 as it relates to the ability of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, and hurricanes; GDC 4 as it relates to structures housing the system and the system itself being capable of withstanding the effects of external missiles; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; GDC 44 as it relates to

- (a) the systems ability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions
- (b) suitable redundancy of components so that safety functions can be performed assuming a single active failure coincident with a loss of offsite power and failure coincident with a loss of offsite power and
- (c) the systems ability to isolate components, systems, or piping so that the systems safety function will not be comprised;

GDC 45 as it relates to allowance for periodic inspection of safety-related components and equipment; GDC 46 as

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it relates to allowance for operational functional testing of safety-related systems or components to assure structural integrity and system leak tightness, operability, and adequate performance of active system components, and the capability of the integrated system to perform the required functions during normal, shutdown, and accident conditions; GDC 61 as it relates to the system design for fuel storage and handling of radioactive materials, including

- (a) capability for periodic testing of components important to safety
- (b) provisions for containment
- (c) provisions for decay heat removal
- (d) capability to prevent reduction in fuel storage coolant inventory under accident conditions in accordance with position C.6 of RG 1.13 and
- (e) capability and capacity to remove fission products, radioactive materials, and impurities from the pool water and reduce occupational exposures;

GDC 63 as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions; and 10 CFR Part 20, paragraph 20.1 (c) as it relates to radiation doses being kept as low as reasonably achievable (ALARA).

Compliance with the requirements of GDC 2 is based on complying with the guidance of RG 1.13, Positions C.1, C.2, C.6, and C.8 as well as RG 1.29, Position C.1 for safety-related portions of system, and Position C.2 for non-safety-related portions of the system. Compliance with the requirements of GDC 4 is based on complying with the guidance of RG 1.13, Position C.2. Compliance with the requirements of GDC 44 is based on the recommendations of Branch Technical Position (BTP) ASB-9-2 for calculating the heat loads and the assumptions set forth in item 1.h of subsection III of SRP 9.1.3 and the pool temperature limitations identified in item 1.d of subsection III of SRP 9.1.3. Compliance with the requirements of 10 CFR Part 20, paragraph 20.1 (c) is based on complying with the guidance of RG 8.8, Positions C.2(f)(2) and C.2.f(3).

The pool cooling and purification system is designed to remove the decay heat produced by spent-fuel assemblies from a unit refueling, complete core off-load, and the accumulated assemblies resulting from previous refuelings.

Of the two independent cooling divisions, each is capable of maintaining the spent-fuel-pool bulk water temperature at or below 49 °C (120 °F) during normal operation. The two cooling divisions cool the water in the spent-fuel pool. Each division has a cooling pump, skimmer, and heat exchanger. The two purification divisions clarify and purify the water in the spent-fuel pool, transfer canal, and refueling pool. Each purification division has a strainer, pump, filter, and demineralizer.

In CESSAR-DC Section 9.1.3.3, ABB-CE presents a safety evaluation, stating that the backup cooling division protects against the loss of one component or loss of cooling water to a heat exchanger. In CESSAR-DC Section 9.1.3.3.2, ABB-CE discusses design protection against the loss of water to the spent-fuel pool. ABB-CE states that gravity cannot be the mechanism for draining the spent-fuel pool because the cooling pumps take suction near the normal water level and the return lines have antisiphon holes. In CESSAR-DC Figure 9.1-3, ABB-CE also shows antisiphon holes in both the suction and discharge lines of the refueling pool purification system, the other system that takes suction on the pool. The spent-fuel pool is separated from the transfer canal by a gate that allows the transfer canal to be drained while maintaining water in the spent-fuel pool. In CESSAR-DC Section 9.1.3.3.3, ABB-CE states that the design flow rates and filtering capabilities of the purification system will be sufficient to meet the stated water chemistry and clarity limitations.

In CESSAR-DC Table 3.2-1, ABB-CE states that the system is located in the nuclear annex building and the steel containment. These structures are designed to seismic Category I requirements which ensure that safety-related portions of the system are protected from earthquakes, external floods, hurricanes, tornados, and missiles generated by these tornados. The spent-fuel pool cooling portion of the system is designed to seismic Category I, Safety Class 3 requirements. The purification portion of the system is classified as non-safety-related with the exception of the penetrations to and from the refueling pool (located inside the containment). These penetrations consist of inboard and outboard manual containment isolation valves (CIVs) on both the suction and discharge lines. The penetrations are designed to seismic Category I, Safety Class 2 requirements. The non-safety-related portion of PCPS is designed so that should it fail, it will not interfere with the safety-related portions of the system.

The safety-related portions of the system are redundant and physically and electrically separated to ensure that the failure of one division will not interfere with the ability of the redundant division to perform its safety function.

The system is designed so that no pipe failure can result in draining the spent-fuel pool below the level required for safe shielding and fuel assembly cooling.

Both spent-fuel cooling and pool-purification piping enter the pool near the normal water level. The spent-fuel cooling return line contains antisiphon devices to prevent pool drainage while the purification portion contains both antisiphon devices and isolation valves to prevent drainage. The pool is separated from the fuel transfer canal by a gate that allows water to remain in the spent-fuel pool when the transfer canal is drained.

Makeup water for the PCPS system is provided by the chemical and volume control system (CVCS), a non-safety-related, seismic Category I, Safety Class 3 system which provides up to 379 L/minute (100 gpm) of borated makeup water to the PCPS. Non-borated makeup water is provided by the demineralized water makeup system (DWMS) to replace evaporative water loss.

Ventilation in the spent-fuel pool area is provided by the FBVS, a safety-related, seismically qualified ventilation system which provides a controlled, monitored release pathway for gaseous effluent from the fuel building portion of the nuclear annex, as well as environmental control for equipment operation. The FBVS is aligned to the filtration mode before any fuel is moved. In addition, the FBVS switches to the filtered mode of operation and initiates a control room alarm when high levels of radiation are sensed upstream of the FBVS filters. System exhaust is monitored before it is released to the environment.

In CESSAR-DC Figure 1.2-5A, ABB-CE shows a reinforced-concrete wall which separates the two divisions of spent-fuel pool cooling, thus, providing adequate divisional separation to ensure that should one division fail because of flooding, fire, missiles, or pipe ruptures, the other division is available to perform the system's cooling function.

On the basis of this information, the staff concludes that the PCPS design complies with the guidelines of RG 1.13, Positions C.1, C.2, C.6, and C.8, as they relate to system and structural design classifications and pool water inventory control, and RG 1.29, Positions C.1 and C.2, as they relate to the design and classification of both safety-related and non-safety-related SSCs. As a result, the PCPS design conforms to the requirements of GDC 2 as it relates to protection against natural phenomena and GDC 4 as it relates to protection against environmental and dynamic effects.

Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section

1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the combined license (COL) applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. This was designated as DSER COL Action Item 9.1.3-1. Upon further review, the staff has determined that the design described in CESSAR-DC does not share SSCs with other nuclear power units. Therefore, the spent-fuel PCPS complies with the requirements of GDC 5, and COL Action Item 9.1.3-1 is acceptable.

The PCPS has features to detect system leakage (including intersystem leakage) and to isolate portions of the system to allow the system to continue to function should portions of the system fail or require maintenance.

Pool leakage is detected by monitoring the following:

- equipment and floor drain sump levels
- CVCS equipment drain tank levels
- area radiation
- spent-fuel-pool levels
- refueling-pool levels

The safety-related portions of the PCPS are redundant, consisting of two independent spent-fuel-pool cooling trains and two independent purification trains. Every component and header can be individually isolated. The independent trains can be cross-connected to ensure that the system can continue to function if parts of the system have to be isolated.

In the DSER, the staff determined that the PCPS complies with the requirements of GDC 44 subject to the resolution of the following open items:

- (1) ABB-CE's response to RAI Q410.56 regarding calculations of heat-generation rates was incomplete. The staff asked ABB-CE to show how the PCPS complies with the assumed conditions given in SRP Section 9.1.3, III H.iii and III H.iv. This was designated as DSER Open Item 9.1.3-1.

Subsequently, ABB-CE prepared Amendments I and L to CESSAR-DC Section 9.1.3.1.4 which state that for normal heat load conditions in the spent-fuel pool (10

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full-power years of spent fuel in the pool along with the newest spent-fuel load in the pool at 120 hours after shutdown), the maximum bulk temperature in the pool is 49 °C (120 °F). A single active failure in the system under these heat load conditions will result in a maximum bulk pool temperature of 60 °C (140 °F). Under maximum heat load conditions (which includes the normal heat load plus a full-core offload 120 hours after shutdown), the bulk pool temperature will not exceed 60 °C (140 °F). With a single active failure in the system under these circumstances, the bulk pool temperature will not exceed 82 °C (180 °F).

On the basis of this additional information, the staff has determined that the heat loads and temperature limits used in the heat generation rate calculations conform to the assumptions outlined in the SRP; therefore, DSER Open Item 9.1.3-1 is resolved.

- (2) ABB-CE's response to RAI Q410.55(a) was inadequate. Although detailed final design information was not available, ABB-CE was asked to submit, as a minimum, the heat removal rates required to comply with the design-basis criteria for normal and abnormal conditions and the design heat removal rates for these conditions. This was designated as DSER Open Item 9.1.3-2.

Subsequently, ABB-CE submitted Amendments I and L to CESSAR-DC Section 9.1.3.1.4, stating that the heat load in the spent-fuel pool under normal heat load conditions is approximately 5.57 MW (19×10^6 BTU/hour), while the heat load in the pool under maximum heat load conditions is approximately 19.6 MW (67×10^6 BTU/hour). On the basis of this additional information, the staff has determined that the heat removal rates have been adequately identified; therefore, DSER Open Item 9.1.3-2 is resolved.

The staff reviewed the security considerations of the spent-fuel PCPS and concluded that the protection provided against draining the pool was inconsistent with Section 2.3.1 of Chapter 7 of Electric Power Research Institute's (EPRI's) ALWR Utilities Requirement Document (URD). EPRI specifies compression seals rather than inflatable seals for fuel-pool gate seals. Also, in Sections 2.3.2.7 and 6.3.1.8.4 of Chapter 4 of the URD, EPRI requires a permanent seal between the reactor pressure vessel and the refueling canal floor that will not be susceptible to any single failure that could cause a rapid draining of the refueling pool. These measures would limit the opportunities during fuel transfer for draining the water that provides radiation shielding and fuel cooling. The staff concluded that the System 80+ design was different from the URD specifications, and that ABB-CE should submit

an explanation. This was designated as DSER Open Item 9.1.3-3.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Section 9.1.4.2.2.13, which clarified the design and function of the refueling-pool seal. This seal connects the upper flange of the reactor pressure vessel to the floor of the refueling cavity in preparation for cavity flooding. The seal is designed to seismic Category I requirements and can withstand various loads, including that associated with a fuel assembly drop from its maximum height while the fuel assembly is being transported by the refueling machine.

The polar crane in the reactor building transports heavy loads (e.g., reactor vessel closure head, upper guide structure assembly) over the refueling pool seal. The crane conforms to the guidelines of NUREG-0612, Section 5.1. These include development and use of safe load paths to transport heavy loads, development and use of operator training and load handling procedures, and provisions for electrical and mechanical stops to minimize the movement of heavy loads over critical components. In addition, administrative controls are used which require that

- no fuel-handling operations be performed while heavy loads are being moved
- the transfer tube valve be closed or the gate between the spent-fuel pool and the fuel-transfer canal be in place and closed before heavy loads are moved over the seal. This ensures that, should the heavy load drop and damage the seal, water will not be drained from the spent-fuel pool.

All seal welds required for structural or sealing integrity are inspectable.

Should the worst-case seal leak occur while transporting a fuel assembly, the refueling machine can move the fuel to a safe position in sufficient time to ensure maintenance of an adequate water cover over the assembly assuming a worst-case seal leak.

On the basis of these design features, the staff concludes that the refueling pool seal is adequately protected from excessive loads and can provide an adequate seal for the refueling cavity. On this basis, DSER Open Item 9.1.3-3 is resolved.

On the basis of its evaluation of the new information in the CESSAR-DC, the staff has determined that the PCPS complies with the requirements of GDC 44 regarding the ability of the system to reliably transfer heat to the ultimate heat sink (UHS), including providing features to ensure system redundancy and isolability.

In addition to the open items discussed above, the staff also identified the following open items in the DSER:

- (1) ABB-CE was asked to submit information showing the effect on criticality of adding non-borated makeup to the pool. This was designated as DSER Open Item 9.1.3-4.

ABB-CE submitted Amendment Q to CESSAR-DC Section 9.1.3.1.4 which clarified that the following design and operational features ensure that the addition of nonborated water will not affect criticality:

- (a) The required boron concentration of 4000-4400 ppm provides sufficient criticality margin.
- (b) The DWMS is isolated by two manual valves that are locked closed and that are opened only as part of an administrative procedure.
- (c) Instrumentation is provided to initiate alarms on high pool level.
- (d) Regular samples are taken to determine boron concentration.

On the basis of this information, the staff has determined that adequate design and operational features ensure that boron dilution beyond safe levels will not occur. On this basis, DSER Open Item 9.1.3-4 is resolved.

- (2) ABB-CE was asked to submit information regarding the effect of equipment wetting (flooding, spray, condensation, high humidity) on system operability. This was designated as DSER Open Item 9.1.3-5.

Subsequently, ABB-CE submitted a clarification that the redundant safety-related portions of the PCPS are protected from the effects of equipment wetting by physical separation. A single divisional wall separates the two divisions of safety-related systems, thereby protecting each division from the effects of fire, flood, missiles, and harsh environmental conditions (including those caused by equipment wetting). Therefore, the plant design allows for the loss of one safety-related division while still performing the system safety function. On the basis of this clarifying information, the staff concludes that the PCPS will be able to perform its safety function. On the basis DSER Open Item 9.1.3-5 is resolved.

- (3) ABB-CE was asked to submit the following additional information on the system's containment isolation valves (CIVs): Isolation time, valve type,

inspection and testing requirements, and power supplies. This was designated as DSER Open Item 9.1.3-6.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Section 9.1.3.3.4.1 which stated that the CIVs for the PCPS consist of inboard and outboard manual isolation valves on both the system suction and return lines. The CIVs are of seismic Category I, Safety Class 2 design. On the basis of this additional information, DSER Open Item 9.1.3-6 is resolved.

- (4) ABB-CE was asked to identify all automatic system functions and primary functions, such as isolation on low flow in the system or in one train, and high temperature. This was designated as DSER Open Item 9.1.3-7.

Subsequently, in Amendment Q to CESSAR-DC, ABB-CE submitted a clarification that only two automatic initiation functions are associated with the PCPS. The first is the automatic bypass of the demineralizers on a high influent temperature signal (see CESSAR-DC Section 9.1.3.2.2.5). This function protects the demineralizer resins from high temperature. The second is the automatic switchover of the FBVS to the filter mode on a high area radiation signal. On the basis of this additional information, DSER Open Item 9.1.3-7 is resolved.

- (5) ABB-CE was asked to submit an analysis of the effect of the location of the spent-fuel-pool suction connection on the pumps when the pool is at its minimum level. This was designated as DSER Open Item 9.1.3-8.

Subsequently, in Amendment V to CESSAR-DC, ABB-CE prepared a discussion in CESSAR-DC Section 9.1.3.2.2.1 regarding the maintenance of pump net positive suction head (NPSH) during worst-case pool conditions. The available system NPSH exceeds each pump's required NPSH based on the minimum pool level and the maximum pool temperature of 82 °C (180 °F). On the basis of this additional information, DSER Open Item 9.1.3-8 is resolved.

- (6) ABB-CE was asked to identify in CESSAR-DC Sections 9.1.3.1.4 and 9.1.3.2.1 the borated makeup water source for the pool, as well as the non-safety-related source of non-borated water used to make up for evaporation losses from the pool during normal operation, as discussed in CESSAR-DC Section 9.1.3.2.1. Also, ABB-CE was asked to identify the building housing the system. This was designated as DSER Open Item 9.1.3-9.

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From its review of CESSAR-DC Figure 9.1-3, the staff determined that the CVCS system provides borated makeup water to PCPS while the DWMS provides non-borated water to the PCPS. Finally, as a result of the review of Table 3.2-1, the staff has determined that the PCPS is located in the nuclear annex and steel containment. On the basis of this information, DSER Open Item 9.1.3-9 is resolved.

7. ABB-CE was asked to specify the seismic category and safety class of the purification portion of the system. This was designated as DSER Open Item 9.1.3-10.

From information added in Amendment Q to CESSAR-DC Table 3.2-1, the purification portion of the PCPS is non-safety-related with the exception of the containment penetrations which are seismic Category I, Safety Class 2. On the basis of this information, DSER Open Item 9.1.3-10 is resolved.

In addition to the seven open items above, the staff noted the following two confirmatory items:

- (1) ABB-CE's response to RAI Q410.55(d)(4), regarding the design's capability to process refueling canal water, was acceptable. ABB-CE was asked to incorporate paragraph 2 of its response into CESSAR-DC Section 9.1.3.3.3. This was designated as DSER Confirmatory Item 9.1.3-1.

Subsequently, ABB-CE submitted Amendment I to CESSAR-DC Section 9.1.3.3.3 which stated that a small amount of water is exchanged between the refueling canal and the spent-fuel pool during fuel handling operations. The PCPS capability will ensure that pool clarity will allow an operator on the refueling machine to read fuel assembly identification numbers which are 9.53 mm (3/8 in.) high, 4.76 mm (3/16 in.) wide, and 1.59 mm (1/16 in.) thick. The orientation of piping in the pool will be such that proper circulation of both pool water and canal water will take place. On this basis, DSER Confirmatory Item 9.1.3-1 is resolved.

- (2) ABB-CE was asked to incorporate paragraph 3 of its response to RAI Q410.67 providing a failure modes and effects analysis (FMEA) on the spent-fuel pool cooling portion of PCPS into CESSAR-DC Section 9.1.3.3.1 to show the effect of this analysis. This was designated as DSER Confirmatory Item 9.1.3-2.

Subsequently, in Amendment L to CESSAR-DC, ABB-CE submitted the FMEA and clarifying information in CESSAR-DC Section 9.1.3.3.1 that redundancy

is provided to ensure that spent-fuel cooling function is maintained assuming a single active failure coincident with the loss of onsite or offsite power. On this basis, DSER Confirmatory Item 9.1.3-2 is resolved.

During normal system operation, the PCPS components operate continuously or periodically. Therefore, special operational testing is not needed. Pool water is sampled regularly to check water chemistry. Seismic Category I portions are inspected in accordance with American Society of Mechanical Engineers (ASME), Code Section XI. The system layout is such that all components of the PCPS can be inspected and tested.

The following additional confirmatory items were identified:

- (1) In its response to RAI Q410.59, ABB-CE presented an acceptable insert for CESSAR-DC Section 9.1.3.1.5 discussing leakage detection and isolation capabilities. ABB-CE was asked to incorporate this response into Section 9.1.3.1.5. This was designated as DSER Confirmatory Item 9.1.3-3.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Section 9.1.3.1.5 which discusses the leakage detection and isolation capabilities of the PCPS. The staff evaluated these features earlier in this section as part of PCPS compliance with GDC 44. On the basis of this additional information, DSER Confirmatory Item 9.1.3-3 is resolved.

- (2) In its response to RAI Q410.61, ABB-CE submitted an evaluation that assures that any failure in the non-safety-related portions of the system cannot affect the functional performance of any safety-related components in accordance with SRP Section 9.1.3, Item III-5 guidance. ABB-CE was asked to incorporate that response to RAI Q410.61 in CESSAR-DC Section 9.1.3. This was designated as DSER Confirmatory Item 9.1.3-4.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Sections 9.1.3.3.1 and 9.1.3.3.2 which states that the pool purification portion of the PCPS has independent flow paths and components which are physically separate from the spent-fuel pool cooling portion of the system. Also, the purification lines which penetrate the spent-fuel pool are isolable and arranged so that their failure will not result in pool drainage below the minimum level to ensure adequate heat removal and radiation shielding. On the basis of the addition of this information, DSER Confirmatory Item 9.1.3-4 is resolved.

On the basis of information given above, the staff concludes that the design of the PCPS complies with the requirements of GDC 45 and 46 regarding system inspection and testing.

The purification portion of the PCPS removes fission products and other contaminants from the pool water so that the dose rate at the pool surface is 25 μ Sv/hour (2.5 mrem/hour) or less. The system maintains pool water chemistry and clarity within the following limits:

- pH between 4.5 and 10 at 25 °C (77 °F)
- chlorides < 0.15 ppm
- such optical clarity that the operator on the refueling machine can read fuel assembly identification numbers which are 9.53-mm (3/8-in.) high, 4.76-mm (3/16-in.) wide, and 1.59-mm (1/16-in.) thick.

Grab samples will be taken periodically to monitor the pool and demineralizer effluent for adequate water chemistry. The minimum water level in the pool ensures adequate radiological shielding for protection of personnel in the area of the spent-fuel pool. System design features to ensure adequate water inventory in the pool, adequate heat removal, adequate isolation and containment, and adequate system testing have been reviewed and discussed earlier in this section.

The staff identified the following additional confirmatory items:

- (1) In its response to RAI Q281.34, ABB-CE discussed demineralizer effluent sampling and resin-bed replacement criteria. In the DSER, the staff asked ABB-CE to incorporate changes to CESSAR-DC Sections 9.1.3.3.3 and 9.1.3.2.2.5 in a revision to the CESSAR-DC. This was designated as Confirmatory Item 9.1.3-5.

Subsequently, ABB-CE submitted Amendment J to CESSAR-DC Sections 9.1.3.2.2.5 and 9.1.3.3.3 which incorporated this information. Each demineralizer produces sufficient water quality for unrestricted access to the working areas around the spent-fuel and refueling pools, as well as adequate water clarity to support refueling operations. Flow distributors in the demineralizers prevent channeling of the influent through the resin bed. A retention element on the demineralizer discharge prevents resin from escaping with the effluent. Overtemperature protection for the resin is provided by an automatic bypass of the resin on high temperature. Resin is replaced when any of the following occur:

- breakthrough of cesium, cobalt, chloride, or fluoride
- excessive differential pressure across the resin bed
- exposure to excessive thermal excursions which approach the resin vendor's recommended limits

On the basis of the incorporation of this additional information in the CESSAR-DC, DSER Confirmatory Item 9.1.3-5 is resolved.

- (2) In its response to RAI Q410.55(d)(5), ABB-CE presented an acceptable response to the staff's question concerning features to prevent the inadvertent transfer of spent filter and demineralizer media. In the DSER, the staff asked ABB-CE to incorporate this material into CESSAR-DC Section 9.1.3. This was designated as Confirmatory Item 9.1.3-6.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Sections 9.1.3.2.2.5, 9.1.3.2.2.6, and 9.1.3.2.2.7, which states that retention elements (discussed previously), pool filters, and pool strainers are used to prevent resins from entering the system and to remove particles from the system that may affect pool clarity. On the basis of the incorporation of this information into the CESSAR-DC, DSER Confirmatory Item 9.1.3-6 is resolved.

After evaluating these features, the staff concludes that the PCPS design complies with the requirements of GDC 61 as it relates to testing, containment, decay heat removal, inventory control, corrosion and fission-product removal, and reduction of radiation exposure.

As discussed previously, the PCPS design includes leak detection and isolation capability. In addition, temperature sensors are used to monitor the thermal condition of the pools to ensure adequate heat removal. Area radiation monitors detect any loss of water shielding over the spent-fuel and refueling pools and initiates actuation of the filter mode of the FBVS on detection of high radiation.

On the basis of this design information, the staff concludes that the PCPS has adequate instrumentation and control features to initiate adequate safety actions should they be needed. This conforms to the requirements of GDC 63 and is acceptable.

PCPS design provisions for achieving and maintaining adequate water chemistry, sampling of pool water to monitor water chemistry, purification of pool water, maintaining a minimum safe shielding water level, sensing excessive radiation levels, and initiating automatic protec-

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tive actions provide assurance that occupational exposures will be kept as low as is reasonably achievable (ALARA).

The staff identified the following additional confirmatory items:

- (1) In its response to RAI Q410.06b, ABB-CE described features of the PCPS design that satisfy Positions C.2.f(2) and C.2.F(3) of RG 8.8 as specified above. In the DSER, the staff asked ABB-CE to incorporate this discussion of PCPS design features into CESSAR-DC Section 9.1.3.1. This was designated as DSER Confirmatory Item 9.1.3-7.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Section 9.1.3.1.6 which describes the radiological controls used to minimize area radiation exposure. These include maintenance of a minimum pool water level to ensure radiation exposures are no more than 25 μ Sv/hour (2.5 mrem/hour) at the pool surface, routing of spent resin transfer lines through shielded pipe chases, design of floor drains to ensure complete collection and routing of radioactive liquid, and the design of the FBVS. In addition, all valves in contact with spent-fuel pool water are made of austenitic stainless steel or an equivalent corrosion-resistant material. All piping in contact with spent-fuel pool water is austenitic stainless steel. On the basis of the incorporation of this information into the CESSAR-DC, DSER Confirmatory Item 9.1.3-7 is resolved.

- (2) In its response to RAI Q410.68, ABB-CE submitted an acceptable rewording to CESSAR-DC Section 9.1.3.3.1. In the DSER, the staff asked ABB-CE to incorporate this rewording into CESSAR-DC Section 9.1.3.3.1. This was designated as DSER Confirmatory Item 9.1.3-8.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Section 9.1.3.3.1 which clarifies that the PCPS performs no emergency function. On the basis of this clarification, DSER Confirmatory Item 9.1.3-8 is resolved.

After evaluating the provisions discussed above, the staff concludes that the PCPS complies with the ALARA guidelines in RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable," and, therefore, complies with the requirements of 10 CFR 20.1(c).

The staff identified the following additional confirmatory items:

- (1) In the CESSAR-DC text, ABB-CE did not identify the related Figure 9.1-3. In the DSER, the staff asked that this information be included in the CESSAR-DC. This was designated as DSER Confirmatory Item 9.1.3-9.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Section 9.1.3.2.1 which referenced Figure 9.1-3. On this basis, DSER Confirmatory Item 9.1.3-9 is resolved.

- (2) In response to RAI Q410.68, ABB-CE committed to incorporating a statement into the CESSAR-DC text stating that the system can perform its cooling function during both normal and accident conditions. Incorporation of this information was designated as DSER Confirmatory Item 9.1.3-10.

Subsequently, ABB-CE submitted this statement in Amendment L to CESSAR-DC Section 9.1.3.3.1. On this basis DSER Confirmatory Item 9.1.3-10 is resolved.

On the basis of the evaluation provided above, the staff concludes that the design of the PCPS complies with the requirements of GDC 2, 4, 5, 44, 45, 46, 61, and 63, and 10 CFR 20.1(c) as specified above and conforms to the guidance in SRP Section 9.1.3.

9.1.4 Fuel-Handling Systems

The staff reviewed the fuel handling systems in accordance with SRP Section 9.1.4 for those portions of the system which handle light loads and in accordance with SRP Section 9.1.5 for those portions of the system which handle heavy loads.

Staff acceptance of the light-load-handling portion of the system design is based on the design complying with the requirements of GDC 2 as it relates to the ability of SSCs to withstand the effects of earthquakes; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; GDC 61 as it relates to a radioactivity release as a result of fuel damage and the avoidance of excessive personnel radiation exposure; and GDC 62 as it relates to criticality accidents. In complying with GDC 61 and 62, specific criteria regarding the maximum kinetic energy of any load lighter than a fuel assembly is identified in item 6, subsection III of SRP 9.1.4. Compliance with the requirements of GDC 2 is based on complying with the guidance of RG 1.13, Positions C.1 and C.6 and RG 1.29, Positions C.1 and C.2. Compliance with the requirements of GDC 61 is based on complying with the guidance of RG 1.13,

Position C.3 and ANS 57.1/ANSI-N208. Compliance with the requirements of GDC 62 is based on complying with the guidance of RG 1.13, Position C.3 and ANS 57.1/ANSI N208.

Staff acceptance of the heavy-load-handling portion of the system design is based on the design complying with the requirements of GDC 2 as it relates the ability of SSCs to withstand the effects of earthquakes; GDC 4 as it relates to protection of safety-related equipment from the effects of internally-generated missiles (i.e. dropped loads); GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; and GDC 61 as it relates to the safe handling and storage of fuel. Compliance with the requirements of GDC 2 is based on complying with the guidance of RG 1.13, Positions C.1 and C.6 and RG 1.29, Positions C.1 and C.2. Compliance with the requirements of GDC 4 is based on complying with the guidance of RG 1.13, Positions C.3 and C.5. Other guidelines used in the evaluation of this system include NUREG-0612, ANS 57.1/ANSI N208, and ANS 57.2/ANSI N210.

According to CESSAR-DC Section 9.1.4, the floor drain system (FHS) is designed to handle and store fuel assemblies and control element assemblies (CEAs). Associated with this system is the equipment used for assembly, disassembly, and storage of the reactor closure head and reactor vessel internals. The fuel and control element handling equipment is designed to transport light loads (loads that are equal to or less than the weight of a fuel assembly and its associated handling tool). This equipment includes the refueling machine, fuel transfer equipment, spent-fuel handling machine, CEA change platform, new fuel elevator, and CEA elevator. The FHS also includes equipment to transport heavy loads (loads weighing more than a fuel assembly and its associated handling tool). Heavy-load-handling equipment includes the overhead cranes in the nuclear annex (the cask-handling hoist and new-fuel-handling hoist of the overhead crane in the nuclear annex) and the reactor building (the main and auxiliary hoists of the polar crane in the reactor building). The FHS equipment includes interlocks, travel-limiting features, and other protective devices to ensure that fuel-handling procedures are performed correctly.

The refueling machine is a rectilinear bridge and trolley system with a vertical mast extending down into the refueling water. The machine is used to handle new- and spent-fuel assemblies within the reactor vessel and refueling cavity inside the containment. Electrical interlocks and limit switches on the bridge and trolley drives prevent damage to the fuel assemblies. A positive mechanical up-stop prevents the fuel from being lifted above the minimum safe-shielding water depth.

The fuel-transfer equipment consists of an underwater, cable-driven double transfer carriage that rides on tracks within the transfer tube and upending machines at each end of the fuel transfer tube that move fuel assemblies from a vertical to a horizontal position (or vice versa) using a hydraulic power unit.

The spent-fuel handling machine is a refueling machine modified for use in the nuclear annex. The principal difference between the machines are the longer bridge span of the spent-fuel handling machine and the interlocks to protect equipment during movement in the spent-fuel pool and cask-laydown area.

The CEA change platform places the operator above the CEA to be moved. The CEA handling tool is then lowered, grappled to the CEA, and relocated.

The new-fuel elevator is used to lower new fuel from the operating floor to the spent-fuel pool. The new fuel is loaded into the elevator by means of the new-fuel handling hoist and new-fuel handling tool. It is then transported by the spent-fuel handling machine to a storage area in region I of the spent-fuel racks or directly to the fuel transfer system.

The CEA elevator is used to assemble new CEAs and disassemble irradiated CEAs. Tools to handle the CEAs are supported from the CEA change platform.

The cask-handling hoist is a 136-tonne (150-ton) hoist used to unload and transport shipping containers for new-fuel from the receiving bay to the new-fuel shipping container laydown area. It is also used for moving the empty spent-fuel cask from the receiving bay to the cask storage pit and for returning the loaded cask to the receiving bay. The receiving bay has sufficient area to allow the cask to be upended with the cask transporter locked in place. The hoist accesses the spent-fuel pool area to facilitate building construction and for installing fuel racks. Mechanical stops on the bridge rails prevent the hoist from traveling over the spent-fuel pool after fuel assemblies have been placed in the fuel racks. An interlock is also provided which prevents the hoist from moving over the new-fuel racks.

The new-fuel-handling hoist is a 9.1-t (10-ton) hoist used to move new-fuel from the new-fuel unloading area to the new-fuel storage racks, the new-fuel inspection stand, and the new-fuel elevator. An interlock is provided which prevents the movement of new-fuel over the spent-fuel racks.

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9.1.4.1 Light-Load Handling Equipment

As stated above, the light-load handling equipment consists of the refueling machine, fuel transfer equipment, spent-fuel handling machine, CEA change platform, new-fuel elevator, and CEA elevator. In CESSAR-DC Table 3.2-1, ABB-CE identifies the location of this equipment as the nuclear annex and/or the steel containment. These buildings are safety-related, seismic Category I structures which are designed to withstand the effects of earthquakes, external floods, tornado winds, missiles generated by tornado winds, and hurricanes. The spent-fuel storage facilities, including the spent-fuel pool and spent-fuel racks, are designed to seismic Category I requirements. The equipment itself is primarily non-safety-related and designed to seismic Category I and II requirements to ensure that no fuel or component fails as a result of a seismic event. Equipment that is not seismically qualified is removed from containment or seismically restrained during normal operation. The equipment is designed so that on a load drop, no damage occurs which could result in inadvertent draining of the spent-fuel pool. This is achieved through the use of electrical and mechanical interlocks which restrict the vertical and horizontal movement of fuel.

On the basis of this information, the staff concludes that the design and location of the light-load-handling portion of the FHS precludes the failure of fuel or safety-related equipment as a result of natural phenomena or the failure of non-safety-related equipment. This complies with the guidelines of RG 1.29, Positions C.1 and C.2, and RG 1.13, Positions C.1 and C.6 as specified above, and, therefore, complies with the requirements of GDC 2 regarding protection of SSCs from seismic events.

Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the combined license (COL) applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. This was designated as DSER COL Action Item 9.1.4-1. Upon further review, the staff has determined that the design described in CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the fuel-handling system complies with the requirements of GDC 5, and DSER COL Action Item 9.1.4-1 is not required.

The refueling and spent-fuel handling machines are not designed to be single-failure-proof, which means that they are not designed to hold their loads given a single failure of a component, nor are they designed to move heavy loads. However, ABB-CE has designed these machines to meet the guidelines of Section 5 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." These guidelines include operator training, load-handling procedures, development of safe load paths, interlocks, and load-drop analyses.

The design of the light-load-handling equipment includes features which protect the spent fuel from damage due to excessive loads, including dropped loads. Specifically, any loads that may be transported over the new-or spent-fuel storage racks, the spent-fuel pool, or the fuel-transfer system are limited in both weight and lift height so that if they fall, the impact energy from the fall will not exceed the design impact energy associated with a dropped fuel assembly and its associated handling tool from its maximum height above the fuel racks during normal fuel handling. This is achieved through the use of electrical interlocks and mechanical stops on the refueling and spent-fuel-handling machines and the new-fuel-handling hoist which limit the height to which a load can be lifted while it is passing over the new and spent fuel. Also, the maximum weight to be lifted is limited by hoist load interlocks and/or hoist motor stall torque.

Interlocks on the fuel-transfer system prevent movement of fuel assemblies when such movement may result in excessive loads. These interlocks include a winch overload interlock, fuel-carrier interlock, upender interlocks, and rotational interlocks. The new fuel and CEA elevator interlocks include hoist cable-slack and hoist lockout interlocks. The hoist lockout interlocks are designed to prevent inadvertent placement of a spent-fuel assembly in an elevator. Finally, the CEA change platform has hoist up-stop, overload, and underload interlocks.

The overhead crane in the nuclear annex building has interlocks which prevent moving heavy loads such as a spent-fuel cask over the new- and spent-fuel racks.

Design features are also provided which ensure positive grappling of a fuel assembly. The refueling and spent-fuel-handling machines engage a grapple actuator arm in axial channels which run the length of the fuel-hoist assembly. Therefore, the assembly cannot be uncoupled, even with an inadvertent uncoupling signal. Fuel assemblies are protected from collisions through the use of an anti-collision device on the mast assembly. The machines are designed to hold their loads on a loss of power or a loss of air, or during an SSE. All equipment in the FHS has

features to allow manual manipulation of equipment to move the fuel to a safe position.

The fuel-handling equipment is tested in accordance with NUREG-0612 and Crane Manufacturing Association of America (CMAA) standards. All equipment is run through several complete operational cycles using a dummy fuel assembly of the same weight, center of gravity, size, and geometry as an actual assembly. The equipment is also checked for functionality under maximum load limits, fuel mislocation and misalignment, and static and dynamic load conditions. Hoisting equipment is tested to 125 percent of hoist capacity and fuel-handling tools are tested to 150 percent of capacity. Manual operations under these conditions are also verified. The COL applicant will prepare specific inspection, test, and operating procedures.

These design features (electrical interlocks, mechanical stops, positive grappling, and testing) help ensure that fuel damage, and subsequent criticality and radiation exposure, will not occur as a result of excessive stresses on the fuel which may occur as a result of dropping the fuel or exposing the fuel to excessive loads.

On the basis of this information, the staff concludes that the light-load-handling portion of the FHS complies with the guidelines of RG 1.13, Position C.3, and with ANS 57.1 and, therefore, conforms to the requirements of GDC 61 and 62 as they relate to the prevention of radioactivity release as a result of fuel damage, avoidance of excessive personnel radiation exposure, and prevention of criticality accidents, respectively.

In the DSER, the staff identified the following confirmatory items:

- (1) In its response to RAI Q410.108, ABB-CE submitted information on the radiation levels around the spent-fuel pool and refueling machine. In the DSER, the staff asked ABB-CE to incorporate this information into CESSAR-DC Section 9.1.4. This was designated as DSER Confirmatory Item 9.1.4-2.

Subsequently, ABB-CE submitted Amendments E and L of CESSAR-DC Section 9.1.4.3.2.D which incorporated the required information. Mechanical stops on both the refueling and spent-fuel handling machines ensure that at least 2.74 m (9 ft) of water cover a fuel assembly. This ensures that radiation exposure from the assembly is 25 μ Sv/hour (2.5 mrem/hour) or less in the work area. The incorporation of this information into the CESSAR-DC resolves DSER Confirmatory Item 9.1.4-2.

- (2) ABB-CE should verify that the two-assembly fuel-carrier design for the fuel-transfer carriage includes criticality considerations. In the DSER, the staff asked ABB-CE to incorporate this response into CESSAR-DC Section 9.1.4.2.3.2.A. This was designated as DSER Confirmatory Item 9.1.4-4.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Section 9.1.4.2.1 which clarifies that the two-cavity transfer system fuel carrier is designed to meet the same criticality conditions as the spent-fuel storage racks. Incorporation of this information resolves DSER Confirmatory Item 9.1.4-4.

- (3) In its response to RAI Q410.69, ABB-CE presented information on administrative controls that will restrict operation of the fuel-transfer tube valve during fuel-handling operations. In the DSER, the staff asked ABB-CE to incorporate a statement to this effect, and also to discuss the reasons for this operating restriction in CESSAR-DC Section 9.1.4.2.1.2. This was designated as DSER Confirmatory Item 9.1.4-6.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Section 9.1.4.2.1.2.A which incorporated this information. The transfer system winch overload interlock will stop the fuel carriage from moving through the transfer tube if the carriage contacts a partially closed transfer tube valve. This resolves DSER Confirmatory Item 9.1.4-6.

After reviewing the information in CESSAR-DC, the staff concludes that the design of the light-load-handling portion of the ABB-CE System 80+ fuel handling system complies with the requirements of GDC 2, 5, 61, and 62 as specified above and the guidelines of SRP Section 9.1.4, and, therefore, is acceptable.

9.1.4.2 Heavy-Load Handling Equipment

The polar crane in the reactor building and the overhead crane in the nuclear annex are used to move heavy loads. The polar crane is used to handle the reactor vessel head, reactor vessel internals, and other equipment located in the reactor building. The cask handling hoist and the new-fuel-handling hoist on the overhead crane are used to handle equipment, tools, and fuel assemblies from the receipt of the new-fuel containers to the shipment of the spent-fuel cask. Specifically, the cask-handling hoist unloads and transports new-fuel shipping containers from the receiving bay to the new-fuel shipping container laydown area. It also moves the empty spent-fuel cask from the truck/rail car unloading area to the cask laydown area and moves a full cask to the loading area. The new-

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fuel-handling hoist moves new fuel from the new-fuel laydown area to the new-fuel storage racks, the new-fuel inspection stand, and the new-fuel elevator.

As stated in CESSAR-DC Table 3.2-1, this equipment is housed in the nuclear annex and the steel containment. These buildings are safety-related, seismic Category I structures which are designed to withstand the effects of earthquakes, external floods, tornado winds, missiles generated by tornado winds, and hurricanes. The spent-fuel-pool storage facilities, including the spent-fuel pool and spent-fuel racks, are designed to seismic Category I requirements. The cranes and hoist themselves are non-safety-related and designed to seismic Category II requirements to ensure that no fuel or component fails as a result of a seismic event. The equipment is designed so that on a load drop, no damage occurs which could result in inadvertent draining of the spent-fuel pool or damage to safety-related equipment. This is achieved through the use of electrical and mechanical interlocks which restrict the vertical and horizontal movement of fuel.

On the basis of this information, the staff concludes that the design and location of the heavy-load-handling components precludes the failure of fuel or safety-related equipment as a result of natural phenomena or the failure of non-safety-related equipment. This complies with the guidelines of RG 1.29, Positions C.1 and C.2, and RG 1.13, Positions C.1 and C.6, and, therefore, conforms to the requirements of GDC 2 regarding protection of SSCs from seismic events.

Neither the cranes in the nuclear annex nor the reactor building are single-failure-proof, which means that they are not designed to hold their loads given a single failure of a component. Instead, ABB-CE has designed this equipment to comply with the guidelines in Section 5 of NUREG-0612 and American National Standards Institute (ANSI) N14.6, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 lbs (4,500 kg) or More for Nuclear Materials."

ABB-CE has committed to designing the heavy-load-handling equipment to comply with the other guidelines of Section 5 of NUREG-0612 which include provision for operator training, load handling, inspection and testing procedures, safe load paths, mechanical stops, and interlocks.

The cask-handling hoist has a minimum capacity of 136 t (150 tons). It has both an electrical interlock and a mechanical stop to prevent the spent-fuel shipping cask from traveling over the new-fuel and spent-fuel storage racks. The load path prevents the cask from traveling within 4.57 m (15 ft) of the edge of the spent-fuel pool

while administrative procedures that will be developed by the COL applicant will require that the cask not be lifted above the elevation of the operating floor. This prevents the cask from being moved over the new-fuel racks. During initial plant construction, the cask-handling hoist can move over the spent-fuel pool area to aid in construction activities and to install the spent-fuel racks. Following rack installation and before spent fuel is placed in the racks, mechanical stops will be installed on the bridge rails to prevent the hoist from moving over the spent-fuel-pool area.

The new-fuel-handling hoist has a minimum capacity of 9.1 t (10 tons). The hoist has electrical interlocks to control the transfer path of new-fuel assemblies and to restrict fuel-handling loads by either limiting the hoist motor stall torque or incorporating load-limiting devices. It is mechanically restricted from moving new-fuel over the spent-fuel racks.

The main hoist of the polar crane in the reactor building has a minimum capacity of 204 t (225 tons) for transporting the reactor vessel head and internal components; the auxiliary hoist has a minimum capacity of 22.7 t (25 tons) to move lighter loads. The crane moves the reactor vessel head and internals between the vessel and various storage areas. It has two independent automatic upper and two independent automatic lower hoist limits, overload limits, speed control, and a load path to prevent damage to fuel or safety-related equipment as a result of a heavy load drop.

All equipment in the FHS has features to allow manual manipulation of equipment to move the fuel to a safe position.

The refueling cavity seal is designed to connect the upper flange of the reactor pressure vessel to the floor of the refueling cavity to permit filling of the cavity for fuel-handling activities. The seal is designed to seismic Category I requirements and can withstand maximum hydrostatic and seismic loads as well as a worst-case fuel assembly drop.

Heavy loads are moved over the pool seal during refueling operations. To ensure that water is not lost from the spent fuel pool during a heavy load drop, administrative procedures will be developed by the COL applicant which require that either the fuel transfer tube valve or the gate between the spent fuel pool and the fuel transfer system canal be closed during movement of heavy loads in the reactor building.

The maximum pool seal leakage resulting from a light load drop is 354 m³/hr (93,500 gal/hr). Should a seal leak result from a light load drop while moving fuel, the fuel can be

moved to a safe location within a maximum of four minutes. Should there be no makeup water to the refueling cavity, it takes approximately four hours to drain the cavity down to the vessel flange. Therefore, there is sufficient water coverage over the fuel assembly during its movement to a safe location. However, to minimize the possibility of having a fuel failure as a result of a heavy load drop, the COL applicant will develop administrative procedures stating that no fuel is in the refueling machine when moving heavy loads over the seal.

The spent-fuel cask laydown area is separated from the spent-fuel pool by a recessed gate and a reinforced-concrete wall. During cask-handling operations, the gate is closed, sealed, and locked. The floor of the laydown area is designed to withstand a cask drop from 9.14 m (30 ft) above the floor without damaging the floor. Should a breach develop in the laydown area, it will not result in the loss of spent-fuel pool water since the gate is recessed and opens into the spent-fuel pool. This ensures that, should the cask-laydown area lose water, the water pressure in the pool will keep the gate sealed.

In the DSER, the staff identified the following open item:

In Section 5 of NUREG-0612 (July 1980) the staff issued guidelines for controlling heavy loads at nuclear power plants. In the DSER, the staff asked ABB-CE to commit to comply with the recommendations in Section 5 of NUREG-0612. This was designated as DSER Open Item 9.1.4-1.

Subsequently, the staff has reviewed CESSAR-DC Sections 9.1.4.1.3, 9.1.4.3.1, and 9.1.4.3.3 and concludes that ABB-CE has committed to compliance with the guidelines of NUREG-0612. As stated above, ABB-CE has performed a load drop analysis, has designed the heavy-load-handling equipment, and has provided guidance for the COL applicant to develop administrative controls to ensure that fuel and safety-related equipment will not be damaged by the worst postulated load drop. On this basis, DSER Open Item 9.1.4-1 is resolved.

On the basis of this information, the staff concludes that the design of the heavy-load-handling equipment and the associated administrative procedures comply with the guidelines of RG 1.13, Positions C.3 and C.5, and, therefore, comply with the requirements of GDC 4 as it relates to the protection of fuel and safety-related equipment from the effects of internally generated missiles (i.e., dropped loads).

As stated previously, although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the

independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. This was designated as DSER COL Action Item 9.1.4-1. Upon further review, the staff has determined that the design described in CESSAR-DC does not share SSCs with other nuclear power units. Therefore, the fuel-handling system complies with the requirements of GDC 5, and DSER COL Action Item 9.1.4-1 not required.

The heavy-load-handling equipment is tested in accordance with ANSI N14.6 and the CMAA standards. All equipment is run through several complete operational cycles using dummy loads of the same weight, center of gravity, size, and geometry as an actual load. The equipment is also checked for functionality under maximum load limits and static and dynamic load conditions. Hoisting equipment is tested to 125 percent of hoist capacity and load-handling tools are tested to 150 percent of capacity. Manual operations under these conditions are also verified. The COL applicant will prepare specific inspection, test, and operating procedures.

On the basis of this information, the staff concludes that the design of the heavy-load-handling equipment complies with the requirements of GDC 61 as it relates to the safe handling and storage of fuel.

In the DSER, the staff identified the following confirmatory items:

- (1) In its response to RAI Q410.107, ABB-CE submitted additional information on the design features of the refueling cavity water seal and the effects of a postulated seal failure or leak. In the DSER, the staff asked ABB-CE to incorporate this information into CESSAR-DC Section 9.1.4. This was designated as DSER Confirmatory Item 9.1.4-1.

Subsequently, ABB-CE submitted Amendments L and N to CESSAR-DC Section 9.1.4.2.2.13, which incorporated the required information. This information was evaluated above. On this basis, DSER Confirmatory Item 9.1.4-1 is resolved.

- (2) In its response to RAI Q410.109, ABB-CE submitted information on the conformance of the FHS to GDC 2 and 61. ABB-CE's Inserts 1, 2, and 3 to CESSAR-DC Section 9.1.4.3 are acceptable. In the DSER, the staff asked ABB-CE to correct Insert

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1 to refer to Positions C.3 and C.5 of RG 1.13. This was designated as DSER Confirmatory Item 9.1.4-3.

Subsequently, ABB-CE submitted Amendments J and E for CESSAR-DC Sections 9.1.4.3.2.D and 9.1.4.3.3, respectively. In these sections, as well as in Section 9.1.4.3.1, ABB-CE committed to compliance with these guidelines and requirements. On this basis, DSER Confirmatory Item 9.1.4-3 is resolved.

- (3) In its response to RAI Q410.66, ABB-CE submitted an evaluation of the spent-fuel cask-drop accident. In the DSER, the staff asked ABB-CE to incorporate this information into CESSAR-DC Section 9.1.4.3. This was designated as DSER Confirmatory Item 9.1.4-5.

Subsequently, ABB-CE submitted Amendment Q to CESSAR-DC Section 9.1.4.3.1 which incorporated the required information. In this section, ABB-CE states that the water level is maintained in the spent-fuel pool during a spent-fuel cask-drop accident which damages the cask laydown area. The cask-handling hoist has an interlock to prevent moving the cask over new and spent fuel. On the basis of incorporation of this information into the CESSAR-DC, DSER Confirmatory Item 9.1.4-5 is resolved.

- (4) In its response to RAI Q410.71, ABB-CE gave specific information on items to be included under preoperational checks. In the DSER, the staff asked ABB-CE to incorporate this information into CESSAR-DC Section 9.1.4.4. This was designated as DSER Confirmatory Item 9.1.4-7.

Subsequently, ABB-CE submitted Amendment L to CESSAR-DC Section 9.1.4.4 which provided the required information. On the basis of the incorporation of this information into the CESSAR-DC, DSER Confirmatory Item 9.1.4-7 is resolved.

- (5) In its response to RAI Q410.72, ABB-CE discussed the overhead paths for heavy loads and the presence of interlocks and other mechanisms to prevent damage from such movements. ABB-CE has incorporated this response into CESSAR-DC Section 9.1.4.3.1. In its response to RAI Q410.73, ABB-CE discussed load paths for the reactor vessel closure head, the core support barrel, the upper guide structure, and the safety-related equipment in the vicinity of the load paths susceptible to damage by load-handling accidents. In the DSER, the staff asked ABB-CE to also incorporate its response to

RAI Q410.73 into CESSAR-DC Section 9.1.4.3.1. This was designated as DSER Confirmatory Item 9.1.4-8.

Subsequently, ABB-CE submitted Amendments I and J to CESSAR-DC subsections 9.1.4.3.1 and 9.1.4.6 which supply the required information. The design of the heavy-load-handling equipment includes electrical interlocks and mechanical stops to restrict the movement of heavy loads. In addition, the COL applicant will prepare administrative procedures to ensure that heavy loads are moved along safe load paths using proper operating procedures. On the basis of the incorporation of this information into the CESSAR-DC, DSER Confirmatory Item 9.1.4-8 is resolved.

After its review of the information submitted in the CESSAR-DC up to and including Amendment U, the staff concludes that the design of the heavy-load-handling portion of the ABB-CE System 80+ fuel handling system conforms to the requirements of GDC 2, 4, 5, and 61 as specified above, the guidelines of SRP Section 9.1.5 and, therefore, is acceptable.

On the basis of this review, the staff concludes that the design of the light-load and heavy-load-handling portions of the fuel handling system conform to the guidance of SRP Sections 9.1.4 and 9.1.5 and is, therefore, acceptable.

9.2 Water Systems

9.2.1 Station Service Water System

The staff reviewed the design of the station service water system (SSWS) in accordance with SRP Section 9.2.1, "Station Service Water System."

The SSWS is an open system that takes suction from the UHS and supplies cooling water to the heat exchangers in the component cooling water system (CCWS) that remove heat released by plant systems, structures, and components. The SSWS returns the heated water to the UHS.

The SSWS comprises two separate, redundant, open-loop, safety-related divisions. Each division cools one of two divisions of the CCWS, which in turn cools essential and nonessential auxiliary loads. The SSWS operates at a lower pressure than the CCWS to prevent raw water from contaminating the CCWS. Each division of the SSWS comprises two pumps, two strainers, two sump pumps, and associated piping, valves, controls, and instrumentation.

The SSWS pumps are located in a seismic Category I structure that is protected from floods and tornado mis-

siles. The SSWS pumps, strainers, and traveling screens are seismic Category I, Safety Class 3, Quality Class 1, and the SSWS sump pumps are seismic Category NN, Safety Class NNS, Quality Class 3, as is shown in CESSAR-DC Table 3.2-1. In addition, the SSWS is designed to preclude any adverse interaction with non-seismic systems in the vicinity. Therefore, the design presented in the CESSAR-DC complies with GDC 2 by complying with the guidance of RG 1.29, Position C.1, with respect to its seismic requirements.

All essential SSWS components are fully protected from floods, tornado-missiles, internal missiles, pipe breaks, pipe whip, and jet impingement. In addition, the system is designed to minimize the potential for water hammer by providing for adequate filling and high-point vents. The SSWS is also installed underground or in buildings that will protect it from adverse environmental conditions. In the event of a loss-of-offsite power (LOOP), the SSWS will be shut down and starts again in accordance with the diesel generator (DG) load sequencing. The DG sequencing times are confirmed to be commensurate with SSWS requirements regarding component cooling. Accordingly, the design presented in the CESSAR-DC complies with GDC 4, as related to effect of missiles inside and outside of containment, effects of pipe whip, jets and environmental conditions resulting from high and moderate energy line breaks and dynamic effects associated with flow instabilities and loads (e.g., water hammer) during normal plant operation as well as during upset or accident conditions.

The staff reviewed the design of the SSWS to identify shared systems and components. The two divisions of the SSWS are physically and electrically separate and share no components or systems. Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant would have to apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures Systems and Components," with respect to the capability of shared structures, systems, and components (SSCs) to perform their required safety functions. Upon further review, the staff has determined that the design described in CESSAR-DC does not share SSCs with other nuclear power units and, therefore, complies with the requirements of GDC 5.

Each division can provide for safety-grade shutdown cooling via both pumps for up to 36 hours and post-loss of coolant accident (LOCA) cooling via one pump for up to 30 days. Each of the four identical SSWS pumps (two per

division) can produce 100 percent of the required flow for post-LOCA conditions. During normal operation, only one pump per division is required to be operating. If a low pump discharge signal is received, the second pump in the respective division starts automatically. The pumps are of the vertical centrifugal type and are installed so that they meet the minimum NPSH at the highest expected operating temperature and flow, at the normal water elevation, and assuming the traveling screens are 50 percent clogged. The available NPSH exceeds the required NPSH for worst case water elevations for all operation, flow and temperature conditions. The minimum available NPSH is the smaller of either 25 percent greater than, or 3 m (10 ft) greater than, the required NPSH specified by the pump vendor. (Note: For worst case UHS water elevation, the margins previously specified need not apply.) The pumps have at least a 7-percent margin in head at the pump design point. The head-versus-flow curve is continually rising from the design point to shutoff.

Instrumentation that monitors the SSWS flow, temperature, and system pressure, as well as radiation levels within the SSWS inventory, supports automatic system actuation features and alarms to alert the operator to anomalous operating conditions. These features ensure that the SSWS is properly removing heat from the CCWS and transferring heat to the UHS. In addition, these features detect pipe breaks and related system failures to minimize the resulting adverse consequences and to prompt mitigating actions. As noted earlier, the design comprises two full-capacity divisions, each of which has two redundant trains to provide the necessary cooling. The system is designed to accommodate a single failure in a train and compensates for the postulated single failure via (1) reliance on the redundant train within the division or (2) the two trains of cooling in the other division, or both, depending on the single failure. The staff also reviewed the design to ensure that isolation valves were installed and could be remotely operated to ensure that the system's safety function would not be compromised by a pipe break, a component failure, or a related failure. ABB-CE incorporated adequate isolation and control provisions into the design to protect the system from postulated failures. Therefore, the system complies with the requirements of GDC 44 with respect to cooling water by providing a system to transfer heat from SSCs important to the safety of the UHS.

Radiation monitors downstream of the CCWS heat exchanger discharge monitor the SSWS flow to detect any leakage from the CCWS to the SSWS. Upon detection of radiation at a pre-set level above background radiation, an alarm sounds in the control room. The operator can isolate the affected component cooling water (CCW) heat

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exchanger from the control room by means of motor-operated butterfly valves.

The SSWS is designed and installed to permit in-service inspections and tests in accordance with ASME Code Section XI. The design of redundant trains within a division provides a mechanism for testing individual components and portions of the system on a periodic basis without affecting plant operations. On this basis, the design is in compliance with GDC 45, as related to design provisions to permit inservice inspection of safety-related components and equipment, and GDC 46, as related to design provisions to permit operational functional testing of safety-related systems and components.

In CESSAR-DC, ABB-CE states that the SSWS will be able to perform its function during abnormally low water levels and that steps are taken to prevent organic fouling and inorganic buildups that may degrade system performance. Proposed steps include traveling screens, proper water treatment, and backwash capabilities. This was DSER COL Action Item 9.2.1-1. By CESSAR-DC Amendment L, ABB-CE stated that a safety-grade traveling screen shall be installed downstream of trash racks and upstream of station service water pumps. Further, ABB-CE stated that the screens shall be equipped for periodic cleaning and designed to limit ingestion of biofouling, organics, and debris, consistent with the fouling design limits of the piping system and CCWS heat exchanger and the need to limit any blockage of the pump inlets. On this basis, DSER COL Item 9.2.1-1 is acceptable.

The design of the SSW system complies with GDC 2, 45, and 46 with respect to inservice inspection and testing requirements and protection against natural phenomena for its safety-related portions. The system design complies with the guidelines of Position C.1 of RG 1.29 with respect to seismic requirements for the safety-related and applicable non-safety-related portions of the system. The system complies with GDC 5 with respect to the sharing of system components between units. Further, the system design complies with GDC 4 and 44, with respect to cooling water requirements and protection against internally and externally generated missiles and dynamic effects resulting from postulated piping failures. Therefore, the staff concludes that the system design complies with the applicable acceptance criteria of SRP Section 9.2.1 and is acceptable.

Security Considerations

The description in CESSAR-DC Section 9.2.1.3(D) of the protected location of SSWS components did not address protection from sabotage. In a letter of February 28,

1992, ABB-CE proposed to amend CESSAR-DC Section 9.2.1.1.4 to state that the SSWS pump structure will be a seismic Category I structure within the scope of the COL applicant and to require a COL applicant to have an SSWS pump structure that is in the same "vital protection area as the main plant" and that will "withstand the effects" of sabotage. The staff interpreted that terminology to mean that equipment within the SSWS pump structure will be protected as vital. However, it was not clear whether or not ABB-CE intended to require that there should be only a single protected area rather than two separate ones. Either alternative was acceptable under the NRC's regulations. Whether COL applicants will be given latitude to select more than one protected area, and whether cables and piping connecting two separate areas will be inaccessible outside separate protected areas, was designated as DSER Open Item 9.2.1-1. By CESSAR-DC Amendment L, ABB-CE added the following statement to CESSAR-DC Section 9.2.1.1.4: "The SSWS is designated as a vital area and all SSWS piping and cabling shall be located within the vital protection area that is common to the main plant." By adding this statement, ABB-CE has clarified the plant design. The use of a single protected area is considered to be an acceptable resolution. On this basis, DSER Open Item 9.2.1-1 is resolved.

The proposed description of the SSWS pump structure was not sufficient for the staff to determine that a structure designed to these requirements will provide adequate vital barrier protection. This was designated as DSER COL Action Item 9.2.1-2. By CESSAR-DC Amendment L, ABB-CE stated in Section 9.2.1.1.2 that the COL applicant will submit a detailed description of the SSWS pump structure. With the inclusion of this information into CESSAR-DC, DSER COL Action Item 9.2.1-2 is acceptable.

In Chapter 8, Section 3.3.5.8, of the URD, EPRI requires that attention be paid to access provided for cleaning out trash racks and traveling screens in order to reduce the potential for bypassing the intrusion detection system at intake areas. The staff concluded that ABB-CE should address this URD requirement. Coordination of the SSWS intake design with the security system design was designated as DSER Open Item 9.2.1-2.

By CESSAR-DC Amendment L, ABB-CE added the following paragraph to amplify the system description to CESSAR-DC Section 9.2.1.1.4:

Trash racks shall be provided upstream of the station service water pumps susceptible to damage due to large debris. A safety grade traveling screen shall be installed downstream of the trash racks and upstream of the station service water pumps.

Provisions for physical access to the trash racks and the traveling screens shall be consistent with the design of security barriers and intrusion detection systems. Provisions for physical access and for debris removal shall not provide a potential path for covert penetration into the protected area.

By adding this information, ABB-CE has clarified the plant design. The use of a safety-grade traveling screen which is consistent with the design of security barriers and intrusion detection systems of Chapter 8, Section 3.3.5.8, of the URD requirements resolves the issue. On this basis, DSER Open Item 9.2.1-2 is resolved.

9.2.2 Component Cooling Water System

The staff reviewed the design of the CCWS in accordance with SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems." The staff reviewed the CESSAR-DC and ABB-CE's response to RAI Q410.111 to develop its summary of the design.

The CCWS cools essential and nonessential nuclear components. In conjunction with the SSWS and the UHS, the CCWS is capable of removing sufficient heat from the essential heat exchangers to ensure a safe reactor shutdown and continuous cooling following a postulated accident. The CCWS consists of two separate, independent, redundant, closed-loop, safety-related divisions. Either division is capable of providing 100 percent of the cooling required for a safe reactor shutdown.

Each division of the CCWS includes the following: two heat exchangers, a surge tank, two CCW pumps, a chemical addition tank, a CCW radiation monitor, two sump pumps, a CCW heat exchanger structure sump pump, and related piping, valves, instrumentation, and controls. No cross-connection exists between the two divisions. The CCWS is cooled by the SSWS that removes heat from the tube side of the CCWS heat exchangers. To preclude leakage from the SSWS to the CCWS, the CCWS operates at a higher pressure than the SSWS.

Each CCWS division consists of an essential and nonessential cooling loop.

The essential cooling loop piping and components (e.g., heat exchangers, pumps, and surge tanks) are seismic Category I and Safety Class 3. CIVs and containment penetration piping are seismic Category I and Safety Class 2. The essential portion of the CCWS supplies cooling to the following redundant safety-related components: shutdown cooling heat exchangers, mini-flow heat exchangers, and pump motor coolers; safety injection pump motor coolers; containment spray heat exchangers,

mini-flow heat exchangers, and pump motor coolers; CCW pump motor coolers; spent-fuel pool heat exchangers and pump motor coolers; motor driven emergency feedwater pump motor coolers; DG jacket water coolers; and essential chillers. The nonessential portion of the CCWS supplies cooling to the reactor coolant pump (RCP) motor air coolers, motor oil coolers, oil coolers, seal coolers, and high pressure coolers; letdown heat exchanger; sample heat exchangers; gas stripper; boric acid concentrator; normal chilled water condensers; and instrument air compressors. Nonessential components are supplied CCW by means of non-nuclear safety class cooling loops with the exception of the charging pump motor coolers and miniflow heat exchangers, the instrument air compressors, and the DG engine starting air aftercoolers, which are supplied CCW by means of Safety Class 3 cooling loops. The isolation valves separating seismic Category I portions from the nonseismic portions are Class 2 or 3 (i.e., Quality Group B or C, respectively). In addition, the CCWS is designed to preclude any adverse interaction with nonseismic systems in the vicinity. Therefore, the design presented in the CESSAR-DC complies with GDC 2 with respect to its seismic requirements by virtue of compliance with the guidance of RG 1.29, Position C.1, with respect to safety-related portions of the system, and Position C.2, with respect to non-safety-related portions of the system.

All essential CCWS components are fully protected from floods, tornado missiles, internal missiles, pipe breaks, pipe whip, and jet impingement. The two divisions of the CCWS are physically separated and are routed so as to be protected from environmental conditions that could impair performance. In addition, in responding to Q410.76, ABB-CE stated that the effects of high- and medium-energy pipe breaks are considered in the design of the CCWS. Specifically, the response indicates: "The CCWS safety-related components are designed and protected such that this type of failure would not affect the safety performance of the CCWS." See Section 3.5 of this report. Accordingly, as presented in the CESSAR-DC, the design complies with GDC 4, as related to effect of missiles inside and outside of containment, effects of pipe whip, jets and environmental conditions resulting from high and moderate energy line breaks and dynamic effects associated with flow instabilities and loads (e.g., water hammer) during normal plant operation as well as during upset or accident conditions.

The CCWS is also designed to minimize the potential for water hammer by having vents in all high points and drains in all low points of the system. Vents are located to ensure that the piping is filled with water; this reduces the chances of water hammer after pump startup. Also, valve opening and closing times are selected to minimize the effects of water hammer. Similarly, the surge tanks,

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which represent the system's mechanism for venting, are located at the system's high point to facilitate venting and filling.

The CCWS is powered by the Class 1E auxiliary power system. In the event of an LOOP, DGs provide power to the auxiliary power system. Each of the two DGs is capable of supplying 100 percent of the power required for operating a division of the necessary safety-related equipment. An LOOP would result in the shutdown of the CCWS and its subsequent restart, in accordance with the DG's load sequencing. The sequencing logic ensures that the appropriate CCWS pump is loaded within approximately 10 seconds.

Several valves in the CCWS rely on the compressed air (instrument air) system for operation. In the CCWS, all active safety-related valves dependent on compressed air have safety-grade operators and solenoid valves. In the event of a loss of the compressed air system (e.g., during an LOOP event), the safety-related solenoid valves are vented and the valve would fail in the prescribed fail-safe position. The motor-operated valves in the essential portions of the CCWS are powered from diesel-backed power sources and would be available following a LOOP.

The CCWS comprises two divisions that are spatially and electrically separated and share no components or systems. This design precludes any single event from affecting both systems. Each division is individually capable of providing the requisite heat removal capability to support a reactor shutdown and continued cooling following a design-basis accident (DBA). In addition, the design has redundant trains within each division ensuring that the failure of an individual component or train will not impair the functionality of the CCWS. Although the System 80+ design can be used at either single- or multiple-unit sites, as described in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. Should a multi-unit site be proposed, the COL applicant would have to apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. This was designated as DSER COL Item 9.2.2-1. Upon further review, the staff has determined that the design described in CESSAR-DC does not share SSCs with other nuclear power units. Therefore, the CCWS complies with the requirements of GDC 5, and DSER COL Action Item 9.2.2-1 is resolved.

The redundant divisions also ensure that safety functions can be performed assuming a single active component

failure coincident with the LOOP. Within each division, ABB-CE has designed component redundancy (e.g., two pumps, two heat exchangers) to ensure that a single failure would not typically compromise the heat removal function of a division.

This component redundancy is complemented by the motor-operated valves that isolate an individual division or individual trains within a division. These isolation valves protect the essential components from failures of the non-essential portions of the CCWS. The piping and instrumentation diagrams (P&IDs) clearly identify the class breaks between the essential and non-essential portions of the system and the valves that perform the isolation function. Two valves in series are at each safety/non-safety interface. These valves automatically close on a low-low surge tank level, preventing the loss of the safety portion of the system and also limiting the water that would be released into the building as a result of a failure in the non-safety portion of the system. The isolation times of these valves are adequate to preclude excessive drawdown of the surge tanks.

Inventory losses that result from a failure in the nonessential portions of the CCWS can be compensated for by makeup water from the surge tanks. Makeup water is normally sent to the surge tanks by the DWMS. However, when the DWMS is unavailable, such as during an accident, a backup make-up water line of seismic Category I design can be provided by installing a spool piece to connect the SSWS to the CCWS surge tank. Using the SSWS as a makeup water system provides a backup to the intact division, since in the case of a major leak in one of the CCWS divisions, the affected division is removed from service and the redundant division is there to be used.

Leakage into or out of the CCWS is detected by level monitoring of both surge tanks and sumps, as well as by radiation monitoring. Safety-related instrumentation on the surge tank in each division alerts the operators to high, low, and low-low levels in the tank that indicate system leakage. These monitors are complemented by high-level alarms for the CCWS sumps and the CCWS heat exchanger structure sumps. In addition, radiation monitors located downstream of the CCWS pumps will detect any leakage of radioactive fluids into the CCWS.

The CCWS cools the following RCP support systems: RCP high-pressure cooler; RCP seal coolers; RCP motor coolers; and RCP motor oil coolers. These RCP support systems are part of the nonessential portion of the CCWS. However, the supply and return headers for the RCP support systems do not isolate on a safety injection actuation signal (SIAS) and will be supplied with cooling water following a small-break LOCA with an LOOP in accor-

dance with the requirements of 10 CFR 50.34(f)(1)(iii). Low-flow and high-flow alarms are provided for the various RCP heat exchangers which use CCWS for cooling. These alarms alert the control room to flow anomalies associated with the heat exchangers and ensure timely identification of the cooling problem to protect the RCP pumps.

On the basis of this discussion, the system complies with the requirements of GDC 44 with respect to cooling water by providing a system to transfer heat to the UHS from SSCS important to safety.

Components of the CCWS can be fully tested during normal operation. The redundant trains of equipment within each division provides flexibility in the scheduling and conduct of inspections. In addition, tests to verify proper operation of individual CCWS components can be conducted using installed bypass and recirculation loops. These tests supplement the system level tests by verifying acceptable performance of each active component in the CCWS. The surveillance and testing requirements are discussed in Chapter 16 of this report. Therefore, the system is in compliance with GDC 45, as related to design provisions to permit inservice inspection of safety-related components and equipment.

The staff reviewed seismic design and isolation provisions between essential and non-essential portions of the CCWS and finds them acceptable based on compliance with GDC 2 as indicated below. In addition, the staff verified that the CCWS will cool essential nuclear components during normal, off-normal, and accident conditions. Accordingly, the CCWS complies with GDC 46, as related to design provisions to permit operational functional testing of safety-related systems or components to insure (a) structural integrity and system leak tightness, (b) operability and adequate performance of active system components, and (c) capability of the integrated system to perform required functions during normal shutdown, and accident situations.

In response to RAI Q410.111(b), ABB-CE added dual isolation valves in series where the essential and nonessential portions of the system meet. ABB-CE also noted several sections and tables in the CESSAR-DC that needed to be revised to add reference to the addition of the valves. However, ABB-CE did not submit any reference to these valves (CC-122, CC-123, CC-222, and CC-223) in CESSAR-DC Section 9.2.2.2.5, "Emergency Operation." This was designated as DSER Open Item 9.2.2-1. By CESSAR-DC Amendment R, ABB-CE has added reference to these valves in Section 9.2.2.2.5, "Emergency Operation." On this basis, DSER Open Item 9.2.2-1 is resolved.

The design of the CCWS complies with GDC 45, 46, and 2 with respect to inservice inspection and testing requirements and protection against natural phenomena for its safety-related portions. The system design also complies with the guidelines of Positions C.1 and C.2 of RG 1.29 with respect to seismic requirements for the safety-related and applicable non-safety-related portions of the system. Further, the system design complies with GDC 44 and 4 with respect to cooling water requirements and protection against internally and externally generated missiles and dynamic effects resulting from postulated piping failures. Therefore, the staff concludes that the system design complies with the applicable acceptance criteria of SRP Section 9.2.2 as related to reactor auxiliary CWS that are required for safe shutdown during normal, operational transient and accident conditions and for mitigating the consequences of an accident or preventing the occurrence of an accident and is acceptable.

Security Considerations

The staff considers that the CCWS is a vital system; therefore, as required by 10 CFR 73.55(c), access to all CCWS components, including pumps, piping, valves, heat exchangers, controls, power supplies, and other essential components and auxiliaries, should require passage through two barriers. (Locked security doors controlling access between two adjacent vital areas are not desirable, if access to each vital area is otherwise controlled.) The description in CESSAR-DC Section 9.2.2.3(D) of the protected location of CCWS components did not address protection from sabotage. Vital designation of the CCWS was designated as DSER Open Item 9.2.2-2. By CESSAR-DC Amendment L, ABB-CE added the following statement to CESSAR-DC Section 9.2.2.1.1: "The CCWS is designated as a vital system and, therefore, will be protected from sabotage." By adding this statement ABB-CE has made it clear that the CCWS is a vital system. On this basis, DSER Open Item 9.2.2-2 is resolved.

In a letter dated February 28, 1992, ABB-CE proposed to add a new section to the CESSAR-DC (Section 9.2.2.1.4) which would state that the CCWS heat exchanger structure is a seismic Category I structure within the scope of the COL applicant and would require that applicant to have a CCWS heat exchanger structure that is in the "vital protection area" and that will "withstand the effects" of a sabotage event. The staff interpreted that terminology to mean that access to equipment within the CCWS heat exchanger structure requires passage through both the protected area fence and an additional vital barrier. The proposed vital area designation complies with the NRC Review Guideline 17 criterion that seismic Category I equipment be considered vital equipment; that designation

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is acceptable. However, the proposed description of the CCWS heat exchanger structure ventilation system and its fresh air intakes did not make reference to ventilation barrier guidance of RG 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls," and was not sufficient for the staff to determine that a structure designed to these requirements will adequately protect the vital barrier. This was designated as DSER COL Action Item 9.2.2-2. By CESSAR-DC Amendment Q, ABB-CE added the following statement to CESSAR-DC Section 9.2.2.1.4: "Ventilation barriers for the CCW Heat Exchanger Structure(s) Ventilation Systems are in accordance with the guidance provided in Regulatory Guide 5.65." By adding this statement, ABB-CE has made the plant design clear. Requiring the CCW heat exchanger structure ventilation system barriers to be designed in compliance with the guidance given in RG 5.65 resolves this item and eliminates the need for DSER COL Action Item 9.2.2-2.

9.2.3 Demineralized Water Makeup System

The DWMS supplies filtered demineralized water to the condensate storage system (CSS) and to other systems throughout the plant that require high-quality, non-safety-related, makeup water. The system consists of a demineralizer with cation, anion, and mixed-bed units, a vacuum degasifier, and a demineralized water storage tank. The DWMS does not perform any safety function or accident mitigation and its failure would not reduce the safety of the plant.

The staff reviewed the design and operational requirements of the system and finds that it includes all components and piping associated with the systems. The review has determined the adequacy of the proposed design criteria and design bases for the DWMS, regarding adequate supply of reactor coolant purity water during all conditions of plant operation.

The design of the DWMS is acceptable because it complies with GDC 2, "Design Bases for Protection Against Natural Phenomena," as related to the safety-related portions of the system being capable of withstanding the effects of earthquakes, and GDC 5, "Sharing of Structures, Systems, and Components," in regard to the effect of sharing in multiple-unit facilities, as recommended by SRP Section 9.2.3 regarding the review of the DWMS from the supply connection of the service or municipal water source to the point of discharge.

9.2.4 Potable and Sanitary Water Systems

The Potable and Sanitary Water Systems (PSWSs) consist of a Potable Water System and a Sanitary Drainage

system. It includes all components and piping from the filtered water source to all points of discharge to the sewage facilities. The portions of the PSWS that are within the Reactor Building, Nuclear Annex, Turbine Building, Radwaste Building, and Service Building are within the scope of the Certified Design. Those portions of the PSWS that are not within these buildings are not within the scope of the Certified Design.

The staff reviewed the design requirements for the PSWSs in accordance with SRP Section 9.2.4, "Potable and Sanitary Water Systems."

In CESSAR-DC, ABB-CE states that the PSWSs serve no safety functions and that any malfunction of the systems will not adversely effect any safety-related system. Design provisions are provided to prevent the release of liquid effluent containing radioactive material from contaminating the PSWS by providing no interconnections with systems having the potential for containing radioactive materials (Interface 9.2.4-1). Additionally, where necessary, air gaps protect against the contamination of the potable water system with radioactive effluents (Interface 9.2.4-2). Designs conforming to these requirements conform to the requirements of GDC 60, as it relates to design provisions provided to control the release of liquid effluents containing radioactive material and preventing contamination of the PSWS.

In the DSER, the staff stated that the design of the PSWSs are site dependent and are, therefore, not described in detail in the CESSAR-DC. Specific PSWS designs will be reviewed as part of site-specific applications referencing this design. This was designated as DSER COL Action Item 9.2.4-1. By CESSAR-DC Amendments Q and T, ABB-CE presented additional information about the PSWS and stated that the COL applicant will provide the information on those portions of the PSWS that are out of scope. ABB-CE has stated that the PSWS shall be designed to meet the requirements of GDC 60. Specifically, there shall be no interconnections between the PSWSs and systems having the potential for containing radioactive materials. Additionally, the COL applicant shall ensure that the sewage treatment facility design complies with applicable State and local regulations. The CESSAR-DC has sufficient interface requirements to ensure that plant specific designs for these systems will comply with the requirements of GDC 60 as specified above. Therefore, DSER COL Item 9.2.4-1 is acceptable.

9.2.5 Ultimate Heat Sink

The staff reviewed the design interface requirements for the UHS in accordance with SRP Section 9.2.5, "Ultimate Heat Sink."

The UHS is an out-of-scope item that will be reviewed for each site-specific application. Only a general discussion of the UHS appears in the CESSAR-DC. The UHS is the source of cooling to the SSWS which removes heat from the CCWS. The CCWS removes heat from essential and nonessential reactor auxiliary loads during all modes of plant operation.

The conceptual design of the UHS presented in the CESSAR-DC is a single, passive, independent cooling water pond and includes the SSWS intake and discharge. However, ABB-CE notes that site-specific conditions may necessitate the use of two ponds to satisfy RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants." The pond has makeup water pumps to maintain water level. Water chemistry is maintained by a site-specific water treatment system, and blowdown limits salinity buildup in the pond. The UHS will be designed to operate for the required nominal 30 days following a postulated LOCA without requiring any makeup water to the source, and without requiring any blowdown from the pond salinity control system.

The UHS shall meet seismic Category I requirements. In addition, the function of the UHS will not be lost during or after such natural phenomena as a safe-shutdown earthquake, a tornado, a flood, or a drought. The conceptual design of the UHS presented in CESSAR-DC complies with RG 1.29, Position C.1 in regards to complying with the seismic Category I guidelines, and RG 1.27, Position C.2 with regards to the capability to withstand, (a) the most severe natural phenomena expected at the site, with appropriate ambient conditions, but with no two or more such phenomena occurring simultaneously, (b) site-related events (e.g., transportation accidents, river diversion) that historically have occurred or that may occur during the plant lifetime, (c) reasonably probable combinations of less severe natural phenomena and/or site-related events, and (d) a single failure of manmade structural features, and position C.3 with regard to two sources of water.

As presented in the CESSAR-DC, the design of the UHS indicates that there are no shared systems or components, therefore the design is in compliance with GDC 5 as related to the capability of shared systems and components important to safety being capable of performing required safety functions. In addition, the design of the UHS ensures the continued operability of the system assuming a single failure of a manmade structure.

The UHS has an SSWS inlet temperature that does not exceed the maximum allowable temperature required for removing heat from the CCWS heat exchangers during a DBA concurrent with an LOOP. This heat removal capacity includes heat loads anticipated from the start of

the accident through the post-LOCA 30-day time period assuming the worst-case meteorological conditions for the site and no makeup water to the UHS. The typical CCWS heat load requirements are given in CESSAR-DC Table 9.2.2-3.

For sites with severe winters, the referencing COL applicant will submit an analysis to show that the function of the UHS is not impaired by ice formation. Where required, the COL applicant will include provisions, such as warm water recirculation, for de-icing the intake structures to prevent flow blockage. This was designated as DSER COL Action Item 9.2.5-1. However, since the UHS is an out-of-scope interface system, the staff will review the site-specific system according to the applicable NRC requirements at the COL application stage. Therefore, DSER COL Action Item 9.2.5-1 is not needed.

The interface requirements for the UHS offer adequate guidelines to ensure that the plant-specific design can comply with GDC 2, 5, and 44. Since ABB-CE does not address provisions for periodic inspections and tests of the UHS, the evaluation of compliance with GDC 45 and 46 will be deferred to the site-specific review. This was designated as DSER COL Action Item 9.2.5-2. However, since the UHS is an out-of-scope interface system, the site-specific system will be reviewed according to the applicable NRC requirements at the COL application stage. Therefore, DSER COL Action Item 9.2.5-2 is not needed. Additionally, ABB-CE has stated that samples of water and substrate will be collected annually to determine if biological fouling organisms have populated the water source.

Therefore, the staff concludes that the use of the interface requirements in the CESSAR-DC should allow a plant-specific applicant to design an acceptable UHS that complies the requirements of GDC 2, as related to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods; GDC 5, as related to the capability of shared systems and components important to safety being capable of performing required safety functions; GDC 44, as related to, (a) the capability to transfer heat loads from safety-related SSCs to the heat sink under both normal operating and accident conditions, (b) suitable component redundancy so that safety functions can be performed assuming a single active component failure coincident with loss of off site power, and (c) the capability to isolate components, systems, or piping if required so that safety functions are not compromised; GDC 45, as related to the design provisions to permit inservice inspection of safety-related components and equipment, and GDC 46, as related to design provisions to permit operation fractional testing of safety-related systems or components. These requirements

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are in accordance with the guidance of SRP Section 9.2.5 as related to the UHS as a source of cooling water to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident including LOCA.

Security Considerations

In CESSAR-DC Section 9.2.5.1.3, ABB-CE states that the UHS is a seismic Category I structure outside the scope of design certification. In CESSAR-DC Section 9.2.5.1.3(D) the following statement appears: "The function of the Ultimate Heat Sink shall be demonstrated not to be lost during or after any of the following events: . . . 4. Sabotage." In CESSAR-DC Section 9.2.5.2, ABB-CE states that the UHS may be a cooling pond, a river, a lake, an ocean, or a combination of cooling pond and lake, river, or cooling tower.

In Section 5.2.7.1 of Chapter 9 of the URD, EPRI specifies that the plant layout should avoid, if possible, having portions of the protected area perimeter abutting a body of water or crossing it. ABB-CE did not address this URD provision. This was designated as DSER Open Item 9.2.5-1. By CESSAR-DC Amendment L, ABB-CE added the following statement to CESSAR-DC Section 9.2.5-19: "Water boundaries that form part of the protected area boundary shall be avoided, if at all possible." Requiring the COL applicant to avoid, if possible, designing the protected area perimeter from abutting or crossing the UHS body of water complies with Section 5.2.7.1 of Chapter 9 of the URD document. On this basis, DSER Open Item 9.2.5-1 is resolved.

9.2.6 Condensate Storage System

The staff reviewed the design interface requirements for the condensate storage facilities in accordance with SRP Section 9.2.6, "Condensate Storage Facilities."

The CSS is a source of derated condensate for makeup to the condenser and is one of the condensate sources of startup feedwater for makeup to the steam generators. It also collects and stores condensate from the drains of miscellaneous systems. A minimum of two stainless steel condensate storage tanks (CSTs) are provided, each with a minimum capacity based on the maximum condensate usage during startup (e.g., maximum steam generator blowdown level x startup duration) plus a 100-percent margin.

The CSS is not a safety-related system since the emergency feedwater system (EFWS) is designated as the safety-

grade makeup water source. Accordingly, the CSS is not seismically designed. However, ABB-CE states that leakage from the CSS or failure of the CSTs, or both will not result in unacceptable environmental effects. As noted in ABB-CE's response to RAI Q410.112, the CSTs are located in the yard and are designed in compliance with RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," to preclude the potential for releasing radioactive liquids and flooding of areas surrounding the CST. A seismically designed dike or retention pond prevents runoff in the event of tank overflow/rupture. Therefore, the CSS complies with Position C.2 of RG 1.29 and with GDC 2, as related to structures housing the system and the system itself being capable of withstanding the effects of earthquakes.

In response to RAI Q410.112(a), ABB-CE stated that the only safety-related system the CSS interfaces with is the EFWS. ABB-CE describes the provisions for isolating the CSS from the EFW in CESSAR-DC Section 10.4.9, "EFW System," and illustrates these provisions in CESSAR-DC Figure 10.4.9-1. ABB-CE also committed to adding a flow diagram of the CSS system in the CESSAR-DC. This was designated as DSER Confirmatory Item 9.2.6-1. By CESSAR-DC Amendment Q, ABB-CE submitted a flow diagram which showed the CSS system. Therefore, DSER Confirmatory Item 9.2.6-1 is resolved.

The CSS is not safety related because it does not affect (1) the integrity of the reactor coolant system (RCS) pressure boundary, (2) the capability to achieve and maintain safe shutdown, or (3) the capability to prevent or mitigate the consequences of accidents that could lead to unacceptable offsite radiological exposures. Therefore, compliance with GDC 44, "Cooling Water;" GDC 45, "Inspection of Cooling Water System;" and GDC 46, "Testing of Cooling Water System," stated to be acceptance criteria in SRP Section 9.2.6, is not applicable to the system. Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. Should a multi-unit site be proposed, the COL applicant would have to apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. On this basis, the CSS complies with GDC 5. Although it is not safety related, this system is designed to produce adequate pump flow, CST overflow/drainage diversion to the turbine building sump, material corrosion

resistance, CST water level, and control room instrumentation.

Therefore, the staff concludes that the CSS and its design as a non-safety-related system, complies with the applicable portions of SRP Section 9.2.6 as related to Condensate Storage Facilities provided to serve as a receiver for excess water generated by other systems such as the main condenser hotwell, the liquid radwaste low activity reprocessed condensate, and the makeup water treatment system, and also to serve as the water supply or makeup source for various auxiliary systems.

9.2.7 Refueling Water System

ABB-CE does not designate any unique system as the refueling water system. The functions of filling, draining, and purifying the borated water used to flood the refueling pool are performed using the following systems: containment spray system; CVCS; PCPS; and shutdown cooling system. These systems are reviewed in Section 9.1.3 of this report.

Before the cavity is flooded for refueling operations, the blind flange on the containment end of the fuel transfer tube is removed, allowing the transfer canal to be flooded simultaneously with the refueling cavity. The refueling cavity is flooded by pumping water from the in-containment refueling water storage tank (IRWST) (at a boron concentration of >4000 ppm) via the containment spray pumps to the refueling pool. The refueling cavity is filled until the level matches the spent-fuel pool level. This takes most of the contents of the IRWST and provides a minimum of nine feet of water over the active portion of a raised fuel element. The pool is kept clear by the skid-mounted refueling water filtration system which consists of a submersible pump, cartridge-type filters, hoses, and associated instrumentation. During refueling operations, the contents of the refueling pool are purified by the PCPS.

To drain the refueling cavity, the shutdown cooling pumps take suction on the cavity via the operating shutdown cooling lines, and return the water directly to the IRWST. When the water level in the refueling cavity reaches the reactor vessel flange, the operation is terminated. The remainder of the water is removed using the pool purification pumps and, finally, by a gravity drain line that connects the lowest elevation of the pool directly to the IRWST.

9.2.8 Turbine Building Cooling Water System

The staff reviewed the design of the turbine building cooling water system (TBCWS) in accordance with SRP Section 9.2.2, "Reactor Auxiliary Cooling Water System."

The TBCWS is a single, closed-loop, cooling water system that cools non-safety-related components in the various auxiliary systems in the turbine building. Cooling is effected through heat exchangers; heat is rejected to the turbine building service water system (TBSWS). The system comprises two 100-percent pumps, two 100-percent heat exchangers, a surge tank, a chemical addition tank, piping, valves, and instrumentation and controls. In the DSER, the staff stated that as described in CESSAR-DC Section 9.2.8.2, the system was not in complete agreement with CESSAR-DC Figure 9.2.8-1. The figure did not show the chemical addition tank, instruments, or all of the components cooled by the TBCWS. This was designated as DSER Open Item 9.2.8-1. By CESSAR-DC Amendment L, ABB-CE has revised CESSAR-DC Figure 9.2.8.2 to include the chemical addition tank, instruments and all the components cooled by TBCWS. On this basis DSER Open Item 9.2.8-1 is resolved.

In the DSER, the staff identified as interface requirements that the TBCWS is to be located in a building that does not contain any safety-related systems, structures, or components and that any failure of the system (including failures that could cause flooding) should not lead to the failure of any safety-related structure. This was designated DSER Interface Item 9.2.8-1. By CESSAR-DC Amendment L, ABB-CE revised CESSAR-DC to state that the TBCWS heat exchanger is located in building/yard areas that do not contain any safety-related components. Therefore, DSER Interface Item 9.2.8-1 is withdrawn.

The system will comply with GDC 2 by adhering to the guidance of RG 1.29, Position C.2 for assuring that the non-safety-related portions of the system can withstand the effects of earthquakes without affecting adjacent safety-related systems.

Because the TBCWS is not safety related and does not share boundaries with a safety system, the remaining requirements (GDC 4, 44, 45, and 46) of SRP Section 9.2.2 do not apply. Therefore, the staff concludes that the TBCWS complies with the applicable requirements of SRP Section 9.2.2.

9.2.9 Chilled Water System

The staff reviewed the design of the chilled water system (CWS) in accordance with SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems."

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CWS are designed to provide and distribute a sufficient quantity of chilled water to air handling units (AHUs) in specific areas. The CWS comprises the following two subsystems: an essential chilled water system (ECWS) that provides safety-related heating, ventilation, and air conditioning (HVAC) cooling loads, and a normal chilled water system (NCWS) that provides non-safety-related HVAC cooling loads.

9.2.9.1 Essential Chilled Water System

The ECWS consists of two divisions of equal size. Each division can supply 100 percent of the cooling capacity required to meet system demands during normal and accident conditions. Each division takes electrical power from independent Class 1E power sources and cooling water from the respective CCWS division. The ECWS provides chilled water to the safety-related HVAC cooling loads in the control room, electrical rooms, mechanical rooms, subsphere pump rooms, and penetration room.

The ECWS is located in a flood-protected and tornado-missile-protected seismic Category I structure. The ECWS is designed in accordance with seismic Category I and Class 1E requirements. The ECWS is protected from pipe breaks, pipe whip, tornado missiles, jet impingement, and severe environmental conditions. ABB-CE did not, however, indicate if the system's design considers potential water hammer concerns. This was designated as DSER Open Item 9.2.9.1-1. By CESSAR-DC Amendment Q, ABB-CE stated that the ECWS is designed to minimize the consequences of potential water hammer. On this basis, DSER Open Item 9.2.9.1-1 is resolved.

Additionally, in CESSAR-DC Section 9.2.9.1(B), the reference to "safety-related portions" of the ECWS was inconsistent with the reference to the ECWS as a "safety-related system" in CESSAR-DC Section 9.2.9.2. Neither CESSAR-DC Figure 9.2.9-1 nor the flow diagrams submitted in response to RAI Q410.113 clarified which portions of the ECWS were safety related. This was designated as DSER Open Item 9.2.9.1-2. By CESSAR-DC Amendment Q, ABB-CE identified safety-related components and non-safety-related components of the ECWS separately in Figure 9.2.9-1. On this basis, DSER Open Item 9.2.9.1-2 is resolved.

On the basis of the discussion, the design presented in the CESSAR-DC complies with the seismic requirements of GDC 2, by virtue of complying with the guidance of RG 1.29, Position C.1, regarding safety-related portions of the system, and GDC 4 regarding environmental and dynamic effects.

ABB-CE indicates that the ECWS and the NCWS are indirectly connected through a heat exchanger. As shown in CESSAR-DC Table 3.2-1, the ECWS heat exchanger is designed to seismic Category I, Safety Class 3. The ECWS heat exchanger is designed to allow the NCWS to serve all of the ECWS during periods of normal operation without directly connecting the water pathways. Therefore, the integrity of the safety-related ECWS is not degraded by postulated failures in the NCWS.

The System 80+ design can be used at either single-unit or multiple-unit sites; however, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. Should a multi-unit site be proposed, the COL applicant would have to apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. This was designated as DSER COL Action Item 9.2.9.1-1. Upon further review, the staff finds that the design described in CESSAR-DC does not share SSCs with other nuclear units. Therefore, the ECWS complies with the requirements of GDC 5 and DSER COL Action Item 9.2.9.1-1 is withdrawn.

Each 100-percent-capacity ECWS division consists of a chiller, a heat exchanger, two chilled water pumps, an expansion tank, a chemical addition tank, piping, valves, controls and instrumentation. The ECWS heat exchanger and associated chilled water pump allow the NCWS to supply 100 percent of the normal ECWS loads without directly connecting the water pathways. In CESSAR-DC Section 9.2.9.2.1(E), ABB-CE states that the pump that serves the essential chilled water heat exchanger can serve as a backup for the pump that serves the essential chiller. However, the flow diagram in CESSAR-DC Figure 9.2.9-1 was not sufficient to show how the cross-connect valve between the pump discharge lines would prevent backflow through the secured pump. ABB-CE was asked to submit P&IDs for both the ECWS and NCWS. This was designated as DSER Open Item 9.2.9.1-3. By CESSAR-DC Amendment S, ABB-CE submitted the updated Figure 9.2.9-1 which shows both ECWS and NCWS. On this basis, DSER Open Item 9.2.9.1-3 is resolved and the ECWS system complies with the requirements of GDC 44 with respect to cooling water by providing a system to transfer heat from SSCs important to safety to the UHS.

ABB-CE indicates that the ECWS provides access necessary to support inservice inspections and functional testing of safety-related components and equipment. Therefore, the ECWS complies with GDC 45, as related to design provisions to permit inservice inspection of safety-related

components and equipment, and GDC 46, as related to design provisions to permit operational functional testing of safety-related systems or components to insure (a) structural integrity and system leak tightness, (b) operability and adequate performance of active system components, and (c) capability of the integrated system to perform required functions during normal shutdown, and accident situations.

The design of the ECWS complies with GDC 2 and 4 with respect to protection against natural phenomena, internally and externally generated missiles, and dynamic effects resulting from postulated piping failures. The design also complies with GDC 5, 44, 45, and 46 with respect to shared systems, cooling water requirements, and inservice inspection and testing requirements. Therefore, the staff concludes that the system design adheres to the applicable acceptance criteria of SRP Section 9.2.2.

9.2.9.2 Normal Chilled Water System

The NCWS consists of two division of equal size. Each division can supply 100 percent of the cooling capacity required to meet system demands during normal conditions. The NCWS is a non-safety-related system. However, the containment cooling systems serviced by this system are designed to operate during an LOOP. The power supply to the NCWS pumps and chiller units is automatically transferred to the alternate ac (AAC) power source when normal power is not available. Each division takes cooling water from the respective CCWS division. The NCWS provides chilled water to the non-safety-related HVAC cooling loads in the reactor building, nuclear annex, and radwaste building.

The NCWS is not safety related because it is not required to ensure that (1) the integrity of the RCS pressure boundary is maintained; (2) the capability exists to achieve and maintain safe shutdown; and (3) the ability to prevent or mitigate offsite radiological exposures is present during accidents. Therefore, GDC 44, 45, and 46, identified in SRP Section 9.2.2 as acceptance criteria for safety-related portions of cooling water systems, are not applicable to the NCWS. Each division of the NCWS enters the primary containment through two penetrations: one for the supply line and the other for the return line. Isolation valves and piping for the primary containment penetrations are safety related and are designed to seismic Category I Safety Class 2 and 10 CFR Part 50 (Appendix B) standards.

As stated above, the rest of the system is not safety related. However, the non-safety-related portions of the system whose failure during a seismic event could affect any structure, system, or component important to safety,

are designed in accordance with seismic Category II criteria. On this basis, the staff finds that the design of the NCWS system complies with Positions C.1 and C.2 of RG 1.29, as addressed by the SRP Section 9.2.2 acceptance criteria with respect to the seismic requirements for the safety-related and non-safety-related portions of the system.

By virtue of their location in seismic Category I, flood-protected and tornado-missile-protected structures, the safety-related portions of the system are protected against damage from natural phenomena. Further, all safety-related systems are protected against flooding that may be caused by system failure, as discussed in Section 3.4.1 of this report. Therefore, the system complies with GDC 2 with respect to protection of its safety-related portions against natural phenomena and protection of other safety-related systems against the consequences of failure of the non-seismic portions of the system, as required by SRP Section 9.2.2 acceptance criteria.

Therefore, the staff concludes that the NCWS complies with the applicable acceptance criteria of SRP Section 9.2.2.

9.2.10 Turbine Building Service Water System

The staff reviewed the design of the TBSWS in accordance with SRP Section 9.2.1, "Station Service Water System."

The TBSWS is a non-safety-related system that removes heat from the TBCWS and rejects the heat to the normal heat sink (cooling towers). The system consists of two, 100-percent-capacity, vertical, wet pit pumps (one on standby), that are located in the TBSWS intake structure. Because of possible contamination of the TBCWS through leaks in the various components of the system, the design operating pressure of the TBSWS is lower than the design operating or transient pressures of the TBCWS. Piping and valves in the TBSWS are made of carbon steel and are coated with a suitable corrosion-resistant material. The TBCWS heat exchangers are constructed of corrosion-resistant materials. Normally, one TBSWS pump is running and the other is in standby. The standby pump is automatically started on loss of the operating pump or low discharge header pressure. However, in the DSER, the staff stated that ABB-CE had not submitted information regarding such system parameters as pump flow requirements, system operating pressure, and heat removal requirements. This was designated as DSER Open Item 9.2.10-1. By CESSAR-DC amendments up to and including Amendment L, ABB-CE submitted the specific pump flow, heat removal and operating pressure requirements. On this basis, DSER Open Item 9.2.10-1 is resolved.

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The following design criteria were identified as interface items in the DSER. The TBSWS is located in a building that does not contain any safety-related systems, structures, or components. Any failure of the system (including failures that could cause flooding) should not lead to the failure of any safety-related structure. This was designated as DSER Interface Item 9.2.10-1. By CESSAR-DC amendments up to and including Amendment L, ABB-CE revised CESSAR-DC to state that the TBSWS is located in a building that does not contain any safety-related components. Therefore, DSER Interface Item 9.2.10-1 is withdrawn.

The system complies with GDC 2 by complying with the requirements of RG 1.29, Position C.2 for assuring that the non-safety-related portions of the system withstand the effects of earthquakes without affecting adjacent safety-related systems.

Because the TBSWS is not safety related and does not share boundaries with a safety system, the remaining requirements (GDC 4, 44, 45, and 46) of SRP Section 9.2.2 do not apply. Therefore, the staff concludes that the TBSWS compliance with the applicable requirements of SRP Section 9.2.2

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

The staff reviewed the compressed air systems in accordance with SRP Section 9.3.1. Conformance with the acceptance criteria formed the basis for the evaluation of the compressed air systems with respect to the applicable regulations, specifically: GDC 1 for quality standards, GDC 2 for earthquake resistance, and GDC 5 for the capability of shared systems and components, important to safety, to perform required safety functions.

The staff based its review on ABB-CE's response to RAI Q410.114. In the response, ABB-CE submitted extensive revisions to the CESSAR-DC. This was designated as DSER Confirmatory Item 9.3.1-1. ABB-CE incorporated these revisions in Section 9.3.1 of Amendment J to the CESSAR-DC. The staff finds the revisions acceptable. Therefore, DSER Confirmatory Item 9.3.1-1 is resolved.

The compressed air systems consists of the instrument air, station air, and breathing air systems. The instrument air system supplies clean, oil-free, dried air to all air-operated instrumentation and valves. The station air system supplies compressed air for air-operated tools, for miscellaneous equipment, and for maintenance purposes. The breathing air system supplies clean, oil-free, low-pressure air to various locations in the plant to protect employees against

contamination while they perform certain maintenance and cleaning operations.

The instrument air system consists of four parallel trains of 100-percent-capacity air compressors of oil-free, water-cooled design, an air receiver, and an instrument air dryer connected in series. Each compressor has an intake filter/silencer rated to remove all particles that exceed 5 microns (0.2 mils). Downstream of each air compressor, the instrument air flows into an instrument air receiver that has adequate reserve capacity to allow the standby compressors to be started following a compressor trip. Downstream of the air receivers, the instrument air passes through an instrument air dryer. Each air-dryer is equipped with a coalescing prefilter, an air dryer assembly, and an afterfilter capable of drying the compressed air to a dewpoint of -40 °C (-40 °F) at line pressure and filtering the air of particulates that exceed 1 micron (0.04 mils). The safety-related air operated valves, which utilize instrument air, are tested during preoperation to verify fail safe operation upon a sudden as well as a gradual loss of air pressure. Therefore, the design presented in the CESSAR-DC complies with the guidance of ANSI MC 11.1-1976 (ISA S7.3) as related to supplying clean, dry, oil-free air to safety-related components and the guidance of RG 1.68 as related to the testing of the instrument air system.

On this basis, the staff concludes that the instrument air system complies with the requirements of GDC 1 with respect to systems and components important to safety being designed, fabricated, and tested to quality standards commensurate with the importance of the safety functions to be performed.

Instrument air lines penetrating the containment are equipped with electrically operated isolation valves (outside the containment) and check valves (inside the containment). The compressors are powered from non-safety-related buses, but they can be manually aligned to the non-Class 1E AAC source standby power supply during an LOOP.

The station air system consists of two oil-free, 100-percent-capacity, station air compressors, each consisting of an intercooler, aftercooler, and moisture separators. Downstream of the compressors, the air flows to air receivers and is then dried by one of two redundant station air dryers before it is distributed throughout the plant via station air headers.

The breathing air system consists of two, oil-free, 100-percent-capacity breathing air compressors, each consisting of an intercooler, aftercooler, and moisture separators. Downstream of the compressors, the air flows to air

receivers and then to air purifiers. These purifiers clean the air to ANSI/CGA G-7.1, "Commodity Specification for Air," requirements.

With the exception of the containment penetrations and isolation devices, the compressed air systems (instrument air, breathing air, and station air) are all classified as non-safety-related, non-seismic Quality Group D. The containment penetrations and isolation devices are classified as Safety Class 2, seismic Category I, Quality Group B. ABB-CE stated that "the remaining portion of this system shall be classified as non-seismic with the exception of portions which may interact with safety-related equipment or components. In this case these portions shall be classified as seismic Category II." In the DSER, the staff stated that ABB-CE should address the latter part of its response to state whether or not there are any portions of the system that need to be classified as seismic Category II and, if required, should identify these portions of the system in both the text and in the appropriate figures. This was designated as DSER Open Item 9.3.1-1. Upon further study, the staff determined that CESSAR-DC Section 3.2.1, which describes the portions of all the systems that will be designed as seismic Category II, is acceptable. Therefore, DSER Open Item 9.3.1-1 is resolved. On this basis, the design presented in the CESSAR-DC complies with Regulatory Position C.2 of RG 1.29 as related to the system being capable of withstanding the effects of earthquakes. On the basis of this information, the staff concludes that the instrument air system complies with the requirements of GDC 2 with respect to seismic design.

The compressed air systems are not required to operate to achieve reactor shutdown or to mitigate the consequences of an accident. Pneumatic devices in safety-related systems are designed to fail in the safe position upon a loss of air. Air-operated isolation valves, including CIVs, fail closed on a loss of air; valves in the steam supply line to the EFWS steam-driven pump turbines fail open.

In CESSAR-DC Chapter 10, ABB-CE originally stated that each main steam isolation valve (MSIV) had an air reservoir to reduce the likelihood of inadvertent closure of the valve (the fail-safe position) and subsequent plant trip due to instrument air failures. However, the reservoirs were not shown on either the main steam supply system diagrams or the instrument air diagrams. Correction of this omission was designated as DSER Open Item 9.3.1-2. After the DSER was issued, ABB-CE changed the design of the MSIVs by revising CESSAR-DC Section 10.3.2.3.3.1 in Amendment R to state that the MSIVs are process-flow-medium operated and a gas reservoir is provided for each MSIV to allow valve operation during

periods of low main steam pressure. The staff noted that the revisions to CESSAR-DC Section 10.3.2.3.3.1 in Amendment R were inconsistent with the text in CESSAR-DC Section 9.3.1. In Amendment U to the CESSAR-DC, ABB-CE revised Section 9.3.1 to be consistent with Section 10.3.2.3.3.1. On the basis of this information, the staff concludes that the instrument air system is no longer required for the operation of the MSIVs and that the main steam supply system diagram is no longer required to be revised. Therefore, DSER Open Item 9.3.1-2 is resolved.

ABB-CE indicated the safety-related portions of the compressed air system in revisions to CESSAR-DC Section 9.3.1.3. However, ABB-CE did not submit a revised Figure 9.3.1-1, to indicate safety-related portions of the system, in the additional information submitted. This was designated as DSER Open Item 9.3.1-3. ABB-CE revised Figure 9.3.1-1 to indicate the safety-related portions of the system in Amendment J to the CESSAR-DC. The staff finds the revised figure acceptable. Therefore, DSER Open Item 9.3.1-3 is resolved.

Also, the figures for the breathing air system and the station air system, did not indicate the portions of the systems that include the containment penetrations. This was designated as DSER Open Item 9.3.1-4. CESSAR-DC Figures 9.3.1-2 and 9.3.1-3 were revised to show the containment penetrations and were included in Amendment Q to the CESSAR-DC. The staff has determined that the revised figures are acceptable. Therefore, DSER Open Item 9.3.1-4 is resolved.

Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions. This was designated as DSER COL Action Item 9.3.1-1. Upon further review, the staff finds that the design described in CESSAR-DC does not share SSCs with other nuclear power units. Therefore, the compressed air system complies with the requirements of GDC 5, with respect to the capability of shared systems and components, important to safety, to perform required safety functions, and COL Action Item 9.3.1-1 is acceptable.

On the basis of the above review, the staff concludes that the design of the compressed air system complies with the requirements of GDC 1, 2, and 5 with respect to quality

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standards, seismic design, and sharing of systems and components, and is therefore, acceptable.

9.3.2 Process and Postaccident Sampling Systems

The staff reviewed the process sampling system and the postaccident sampling system described in CESSAR-DC Section 9.3.2 in accordance with SRP Section 9.3.2. The areas reviewed were design objectives, design criteria, TSs in this area, process streams sampled and parameters determined through sampling by both systems, piping and instrumentation drawings, location of sampling points and sample stations, purging of sample lines, seismic design and quality group classifications, isolation, and promptness of obtaining samples under accident conditions. The staff's open items are resolved as follows:

In CESSAR-DC Section 9.3.2.1.1.H, ABB-CE describes post-accident sampling. The post-accident sampling system (PASS) is integrated with the process sampling system. Provision is made for diluting liquid and gaseous samples for subsequent radiological analysis. Samples are collected and diluted remotely, to the maximum extent feasible. Grab samples are used for specific laboratory analyses, and on-line monitoring is used for trending. Gas chromatography is not used for on-line analyses. In response to Q281.43, ABB-CE stated that a backup power source is available that can be energized in time to meet the time limit specified in NUREG-0737 during LOOP.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the staff recommended that the Commission approve its position for evolutionary and passive ALWRs of the pressurized water reactor (PWR) type that they be required to have the capability to analyze for dissolved gases in the reactor coolant and for hydrogen in the containment atmosphere in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item III.B.3 of NUREG-0737. The staff acknowledged that determination of chloride concentrations, although helpful in ensuring that plant personnel take appropriate actions to minimize the likelihood of accelerated primary system corrosion following the accident, is a secondary consideration since long-term samples could likely be taken at a low pressure. It does not constitute, therefore, a mandatory requirement of the PASS. The time for taking these samples can be extended to 24 hours following the accident.

The staff also recommended that the Commission approve the deviation from the requirements of Item II.B.3 of NUREG-0737 with regard to requirements for sampling reactor coolant for boron concentration and activity

measurements using the PASS in evolutionary and passive ALWRs.

The rationale for this is that both these measurements are used only to confirm the accident mitigation measures and conditions of the core obtained by other methods and do not need to be performed in an early phase of an accident. Neutron flux monitoring instrumentation which complies with Category I criteria of RG 1.97, will have fully qualified, redundant channels that monitor neutron flux over the required power range. Therefore, sampling for boron concentration will not be needed for the first eight hours after an accident. Samples for activity measurements provide the information used in evaluating the condition of the core. However, this information will be made available during the accident management phase by monitoring other pertinent variables. Accordingly, sampling for activity measurement could be postponed until 24 hours following an accident.

In its July 21, 1993, Staff Requirements Memorandum (SRM), the Commission approved the staff positions to exempt the PASS for ALWRs of PWR design from determining the concentration of hydrogen in the containment atmosphere in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item III.B.3 of NUREG-0737. It also approved extending the time limit for analysis of the coolant for boron and activity to eight hours and 24 hours, respectively. The Commission modified the staff's recommendations regarding evolutionary and passive ALWRs of the PWR type to have the capability to determine the gross amount of dissolved gases (not necessarily pressurized) as a means to meet the intent of 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737.

On the basis of the Commission's decisions and the staff's review of the System 80+ design, as described below the staff concludes that the special circumstances described in 10 CFR 50.12(a)(ii) exist and an exemption from the requirements of 10 CFR 50.34(f)(s)(viii) is justified.

In Amendment T to CESSAR-DC, ABB-CE submitted additional information to show how the design would comply with the modified requirements of NUREG-0737, Issue II.B.3 and Criteria II.5.a, and II.5.c in SRP Section 9.3.2. The proposed PASS complies with Criterion II.5.b (SRP Section 9.3.2) by not using gas chromatography for online analysis.

The System 80+ design has a PASS that complies with the requirements of 10 CFR 50.34(f)(2)(viii) and Item III.B.3 of NUREG-0737 with the modifications described in SECY-93-087. The system has the capability to analyze coolant for dissolved gases 24 hours following the accident and provide the information needed in evaluating

postaccident conditions existing in the reactor vessel. The gases that are released from the reactor coolant during depressurization, are measured at the sampling point itself, and then the depressurized sample is analyzed in the laboratory. The PASS will be able to sample and analyze for activity in reactor coolant and containment atmosphere 24-hours following the accident. The information needed for evaluating the conditions of the core will be provided by the containment high-range area monitor, the containment hydrogen monitor, and core exit thermocouples during the accident management phase. The need for PASS activity measurements will arise only during the accident recovery phase and the 24-hours sampling time is, therefore, adequate. The PASS will also be able to determine boron concentration in the reactor coolant. It will be capable of making this determination within eight hours following the accident. The concentration of boron is required for providing insights for accident mitigation measures. Immediately after the accident, this information will be obtained by the neutron flux monitoring instrumentation which is designed to comply with the criteria of RG 1.97, and which has fully qualified redundant channels capable of monitoring over the full power range. Boron concentration measurements will not, therefore, be required for the first eight hours after the accident.

The staff concludes that the System 80+ design complies with the Commission-approved requirements discussed above for sampling for boron and activity in the postaccident environment in evolutionary and passive AWLRs and is, therefore, acceptable.

In Amendment R to CESSAR-DC, ABB-CE changed the title of Section 9.3.2 from "Process Sampling System" to "Process and Post-Accident Sampling Systems" to be consistent with SRP Section 9.3.2. On this basis, DSER Open Item 9.3.2-1 is resolved.

In Amendment T to CESSAR-DC, ABB-CE failed to specify pressurized sampling capability for the post-accident sampling system in Table 9.3.2-2. However, ABB-CE modified this table in Amendment U to include a pressurized sample capability for the sampling points in the hot leg and shutdown cooling system miniflow heat exchanger. On this basis, DSER Open Item 9.3.2-3 is resolved.

The process and postaccident sampling systems comprise piping, valves, heat exchangers, and other components associated with the systems from the point of sample withdrawal from a fluid system up to the analyzing station, sampling station, or local sampling point. The staff reviewed the ABB-CE proposal to sample all principal fluid process streams associated with plant operation and ABB-CE's proposed design of these systems. The staff

reviewed descriptive information of sampling points, as shown on piping and instrumentation diagrams.

The staff concludes that the design of the process and postaccident sampling systems is acceptable and that the process sampling system meets the relevant requirements as specified below, of 10 CFR 20.1(c) and GDC 1, 2, 13, 14, 26, 41, 60, 63, and 64 in Appendix A to 10 CFR Part 50 and the clarifications of Item II.B.3 in NUREG-0737, as modified by SECY-93-087 and the associated SRM of July 21, 1993.

The staff has determined that the proposed process sampling system complies with (1) the requirements of GDC 13 to monitor variables that can affect the fission process for normal operation, anticipated operational occurrences, and accident conditions, by sampling the reactor coolant, the ECCS core flooding tank, the refueling water storage tank, the boric acid mix tank, and the boron injection tank for boron concentrations; (2) the requirements of GDC 13 and 14, to monitor variables that can affect the reactor coolant pressure boundary and to assure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture, respectively, by sampling the reactor coolant and the secondary coolant for chemical impurities that can affect the reactor coolant pressure boundary; (3) the requirements of GDC 26, to control the rate of reactivity changes, by sampling the reactor coolant, the refueling water storage tank, and the boric acid mix tank for boron concentration; (4) the requirements of GDC 13, 14, and 41, to monitor variables that can affect the integrity of the reactor core and reactor coolant pressure boundary and to reduce the concentration and quality of fission products released to the environment following postulated accidents, respectively, by sampling the chemical additive tank for chemical additive concentrations, to ensure an adequate supply of chemical for meeting the material compatibility requirements and the elemental iodine removal requirements of the containment spray and recirculation solutions following a postulated accident; and (5) the requirement of GDC 64, to monitor for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents, by sampling the reactor coolant, the pressurizer tank, the steam generator blowdown, the secondary coolant condensate treatment waste, the sump inside the containment, the containment atmosphere, and the gaseous radwaste storage tank for radioactivity.

The staff has further determined that the proposed process sampling system complies with (1) the requirements of 10 CFR 20.1(c) to keep radiation exposures ALARA and of GDC 60 to control the release of radioactive materials to the environment, by purging and draining sample streams back to the system or origin or to an appropriate

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radwaste treatment system, and by providing either redundant isolation valves that fail in the closed position or passive flow restrictions in the sampling lines; and (2) the requirements of GDC 63 to detect conditions that may result in excessive radiation levels in fuel storage and radioactive waste systems, by sampling the spent-fuel pool water and the gaseous radwaste storage tank for radioactivity.

The staff also has determined that the proposed sampling and postaccident sampling systems comply with the quality standards requirements of GDC 1 and the seismic requirements of GDC 2 by designing the sampling lines and components of the process and postaccident sampling systems to conform to the classification of the system to which each sample line and component is connected, in accordance with Positions C.1, C.2, and C.3 of RG 1.26, Positions C.1, C.2, C.3, and C.4 of RG 1.29, and the guidelines of RG 1.97.

In addition, the staff has determined that the proposed postaccident sampling system complies with (1) the requirements of Item II.B.3 in NUREG-0737, as modified by SECY-93-087 and its associated SRM, by providing a sampling program to obtain and promptly analyze samples from the reactor coolant and the containment atmosphere, for radionuclides which may be indicators of the degree of core damage, and for dissolved gases and boron concentrations in liquids, following a postulated accident, in accordance with the guidelines of RG 1.97; and (2) the requirements of GDC 60 to control the release of radioactive materials to the environment, by preventing high-pressure carrier gas used in gas chromatographic analysis of reactor coolant from entering the reactor coolant, and by providing passive flow restrictions or fully qualified, remotely operated isolation valves in the sample lines, to limit potential leakage from the sampling lines.

9.3.3 Equipment and Floor Drainage System

The staff evaluated the equipment and floor drainage system (EFDS) according to the guidance of SRP Section 9.3.3. The EFDS is reviewed to ensure its capability to collect and dispose of all waste liquid effluents so that they will be processed in a controlled and safe manner. The EFDS is evaluated to determine whether it meets the following criteria: GDC 2 relative to safety-related portions of the system being capable of withstanding the effects of earthquakes; GDC 4 relative to the system's capability to withstand the effects of flooding and the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and GDC 60 relative to providing a means to control suitably the release of radioactive materials in liquid effluent, including anticipated operational occurrences.

The DG building sump pump system is discussed in Section 9.5.9 of this report.

The equipment and floor drainage system includes equipment and floor drains, vertical drain risers, sloped horizontal drain pipes, sumps, sump pumps, piping, valves, controls, and instrumentation needed to carry radioactive and nonradioactive liquids to the appropriate collection or disposal point. Most of the systems are not safety related, that is Quality Group D and non-seismic. Piping whose failure could flood areas where safety-related equipment is located will be designed to seismic Category I requirements. The containment penetrations and the isolation valves for the containment building drain system are Safety Class 2, Quality Group B, and seismic Category I. Therefore, the design presented in the CESSAR-DC complies with Regulatory Positions C.1, and C.2 of RG 1.29 as related to the system being capable of withstanding the effects of earthquakes. On this basis, the staff concludes that the EFDS complies with the requirements of GDC 2 with respect to seismic design.

The sumps are sized to accommodate drainage required during normal plant operations and maintenance activities. Sump pump capacities are sized large enough to handle the maximum leakage rate into their respective sumps. Potential discharge of fixed fire suppression systems and fire hoses is considered in the sizing of the floor drains. Portions of the reactor building subsphere floor drainage system protect some ESF equipment from floods. The sump pumps in these areas (two 100-percent-capacity pumps for each of the four sumps powered from Class 1E power supplies) are designated as safety related. One sump is located in each of the four quadrants of the subsphere, which are separated by walls containing no unsealed penetrations up to elevation 70+0. Backflow check valves in the EFDS prevent backflow into the safety-related equipment area. On this basis, the staff concludes that the EFDS complies with the requirements of GDC 4 with respect to withstanding the effects of flooding and the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

The EFDS segregates and transports liquids containing wastes to the liquid waste management system (LWMS). Non-contaminated floor drain waste is routed to the turbine building floor drain sump where it is monitored for radioactivity and released to the environs. The equipment and floor drains are separated into equipment drains, floor drains, chemical waste drains, and detergent waste drains. Liquid wastes are routed to the LWMS subsystem that processes the particular waste type.

Separate drain headers are provided for each drain type to prevent mixing of different types of liquids. To prevent

inadvertent release of radioactive waste to the environment, the drainage and collection systems used to handle radioactive or potentially radioactive liquid waste are separate and isolated from the systems used to handle strictly non-radioactive waste. The exception to this requirement is the turbine building floor drain sump. That sump normally handles non-contaminated waste, but it is equipped with a continuous radiation monitor and a normally isolated cross-connection for routing radioactive liquid wastes to the LWMS. Fail-safe air-operated valves (AOVs) prevent inadvertent release of contaminated waste to the environment. When the radiation monitor detects radioactivity in the sump discharge, the sump discharge AOV automatically closes and an alarm automatically sounds to alert the operator. The contaminated waste is then routed to the LWMS for processing by the operator. A loss of air to the AOVs causes the valves to fail in a safe position with the sump discharge isolated and the cross-connection to the LWMS closed. On this basis, the staff concludes that the system complies with the requirements of GDC 60 with respect to preventing the inadvertent transfer of contaminated fluids to a non-contaminated drainage system for disposal.

In CESSAR-DC Section 9.3.3.2.2, ABB-CE notes that the sump pumps in the reactor building subsphere floor drainage system are safety grade, and in CESSAR-DC Table 3.2-1 their classification is given as seismic Category I Safety Class 3. ABB-CE committed to incorporate this information in the text and figures of CESSAR-DC Section 9.3.3. Also, the check valves which provide backflow protection for safety-related equipment should be identified as being seismic Category I, Safety Class 3. ABB-CE's commitment to incorporate this information in CESSAR-DC Section 9.3.3 was designated as DSER Confirmatory Item 9.3.3-1. ABB-CE revised Section 9.3.3.2.2 and Figure 9.3.3-2 in Amendment J to the CESSAR-DC to include the safety grade of the sump pumps and the backflow check valves. Upon further evaluation, the staff determined that the intent of the confirmatory item is met since the sump pumps are classified in CESSAR-DC Table 3.2-1. Therefore, Confirmatory Item 9.3.3-1 is resolved.

Additionally, figures in CESSAR-DC Section 9.3.3 did not indicate where system component safety classifications change. ABB-CE stated that the P&IDs in CESSAR-DC Section 9.3.3 would be revised as necessary to indicate component safety classification and the check valve designation would be incorporated in a future revision to the CESSAR-DC. This was designated as DSER Confirmatory Item 9.3.3-2. ABB-CE revised Figures 9.3.3-1 through 9.3.3-5 to indicate the safety classification of the piping and components, including the backflow check valves, in Amendment J to the CESSAR-DC. The staff

finds the revised figures acceptable. Therefore, DSER Confirmatory Item 9.3.3-2 is resolved.

The staff noted discrepancies between the text in CESSAR-DC Section 9.3.3 and the designated inputs to the floor drain waste header and tanks in CESSAR-DC Figure 11.2-1. ABB-CE's commitment to revise Figure 11.2-1 as necessary, was designated as DSER Confirmatory Item 9.3.3-3. ABB-CE revised Figure 11.2-1 to agree with the text in Section 9.3.3 in Amendment J to the CESSAR-DC. However, after Amendment J was submitted, the staff found additional discrepancies between the text in Section 9.3.3 and Figure 11.2-1. In Amendment U to the CESSAR-DC, ABB-CE revised the text in Section 9.3.3 to agree with Figure 11.2-1. Therefore, DSER Confirmatory Item 9.3.3-3 is resolved.

On the basis of the above review, the staff concludes that the design of the equipment and floor drainage system complies with the requirements of GDC 2, 4, and 60 with respect to seismic design, environmental conditions, and control release of radioactive materials, and is, therefore, acceptable.

9.3.4 Chemical and Volume Control System (Including Boron Recovery System)

The main function of the CVCS is to control the primary coolant inventory and its chemistry, including soluble neutron absorber (boron) concentration, and to maintain seal water injection flow to the RCPs. In the System 80+ standard design, the CVCS is not required to perform any accident mitigation or safe-shutdown functions that are performed by other dedicated safety systems. The system is, therefore, designed as a non-safety-related system. However, since its proper operation is essential for the normal day-to-day operation of the plant, the CVCS is designed with a high degree of reliability and redundancy.

Two centrifugal charging pumps send borated water to the RCS and reactor coolant seals. The volume control tank provides surge capacity for that part of the reactor coolant not accommodated by the pressurizer following load transients. It also provides a means to introduce different additives to the reactor coolant for controlling oxygen concentration and pH and to remove impurities from the primary coolant. Gaseous fission products in the volume control tank that come from the letdown line are removed by a gas stripping system.

Boron concentration changes in the reactor coolant during normal operations are controlled through makeup water and feed and bleed. Boric acid solution of approximately 12.5 percent by weight is prepared in the boric acid batching tank. The solution is diluted to 2.5 percent by

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weight and stored in the boric acid storage tank. All the lines carrying high-concentration boric acid solution are heat traced.

The CVCS purifies the primary coolant by passing letdown flow through heat exchangers and purification ion exchangers. Mixed-bed demineralizers are used to remove ionic and particulate impurities. There is also a deborating ion exchanger capable of reducing the boron concentration in the reactor coolant from 30 ppm to 0 ppm using two charges of ion resin. Instrumentation isolates the ion exchangers on high letdown temperature.

The excess of water from the volume control tank and the water collected in the reactor drain and equipment drain tanks is diverted to the pre-holdup ion exchanger. The purified water passes through the gas stripper and is collected in the holdup tank (HUT). When a sufficient volume accumulates in the HUT, it is transferred to the boric acid concentrator where its boric acid concentration increases to approximately 2.5 percent by weight. This solution is pumped directly to the boric acid storage tank.

The staff reviewed the CVCS in accordance with SRP Section 9.3.4. On the basis of its review of ABB-CE's proposed design criteria, design bases, and safety classification for the CVCS, and the requirements for system performance, the staff concludes that the design of the CVCS including the boron recovery system, is acceptable because it complies with the following requirements:

- GDC 1 and the guidelines of RG 1.26 by assigning quality group classifications to system components in accordance with the importance of the safety function to be performed
- GDC 14 by maintaining reactor coolant purity and material compatibility to reduce corrosion and, thus, reduce the probability of abnormal leakage, rapid propagating failure, or gross rupture of the reactor coolant pressure boundary
- GDC 29 as related to the reliability of the CVCS to provide negative reactivity to the reactor by supplying boric acid water to the RCS in the event of anticipated operational occurrences
- GDC 60 and 61 with respect to confining radioactivity by venting and collecting drainage from CVCS components through closed systems

The CVCS does not need to comply with the requirements of the following GDC, as specified in the SRP, because it has no safety-related functions assigned to it and it is not credited for RCS inventory control:

- GDC 2 related to protection against natural phenomena
- GDC 5 related to sharing of SSCs
- GDC 33 related to reactor coolant makeup
- GDC 35 related to emergency core cooling

In the DSER, the staff asked that ABB-CE provide its basis for classifying the RCP seal injection function of the CVCS as non-safety related. This was designated as DSER Open Item 9.3.4-1.

Subsequently, ABB-CE added an additional system, the dedicated seal injection system (DSIS), to provide RCP seal injection. They also provided clarification on the classification of the system. As discussed in more detail in section 3.2.2 of this report, the staff's concerns have been addressed. DSER Open Item 9.3.4-1 is, therefore, resolved.

Security Considerations

In CESSAR-DC Section 9.3.4.3.5.1, ABB-CE lists valves that can isolate the RCS from the CVCS piping that is accessible outside the containment. The staff concludes that since the isolation valves inside the containment are all either check valves or fail-safe pneumatic valves, none of the CVCS outside the containment needs to be protected as vital equipment.

9.4 Air Conditioning, Heating, Cooling and Ventilation Systems

In a letter dated December 17, 1991, responding to RAI Q500.19, ABB-CE committed to add a statement to CESSAR-DC Section 9.4 stating that HVAC penetrations through security barriers are designed to meet the intent of RG 5.65, for protection of access to vital areas. This was designated as DSER Confirmatory Item 9.4-1. By CESSAR-DC Amendment J, ABB-CE incorporated a statement that the HVAC penetrations through security barriers are to provide security protection to meet the intent of RG 5.65. With the inclusion of this information, DSER Confirmatory Item 9-4-1 is resolved.

9.4.1 Control-Complex Ventilation System

The staff reviewed the control-complex ventilation system (CCVS) in accordance with SRP Section 9.4.1 (NUREG-0800). The system consists of AHUs with filters, and heating and cooling elements to maintain proper environmental conditions and control contaminant levels.

The CCVS consists of the safety-related main control room air conditioning system (MCRACS), the non-safety-related technical support center air conditioning system (TSCACS) and the balance of control-complex safety-related and non-

safety-related air-handling systems. The CCVS serves the control room emergency zone (CREZ), and such safety-related areas as safety-related electrical rooms, vital instrument and equipment rooms, the remote-shutdown room, and battery rooms. It also serves such non-safety-related areas as non-safety-related electrical rooms, non-safety battery rooms, operations and technical support centers, the computer room, shift assembly and offices, the radiation access control room, and central alarm station security group areas, personnel decontamination rooms, and the break room.

The redundant air intakes for main control room air conditioning system (MCRACS), which also serve the non-safety-related technical support center air conditioning system (TSCACS), and the two air intakes for the balance of control-complex air-handling systems, which serve the safety-related and non-safety-related areas, are located on opposite sides of the building. Each MCRACS outside-air inlet has a louver, a tornado damper to prevent depressurization of the areas served by MCRACS and TSCACS during a tornado, two motor-operated isolation dampers, redundant toxic gas and radiation monitors, and a smoke detector. The control room air intake would be isolated automatically on high toxic gas signal. The system also can be isolated manually from the control room. Filtered air intake and recirculation actuates automatically on a SIAS or upon high radiation signaled from the radiation monitors in either of the control room air intakes. Recirculation may be actuated manually from the control room. Two outside-air intakes for the balance of control-complex air-handling systems serve the safety-related and non-safety-related control areas and subsphere building ventilation system. Each has a louver, a tornado damper which will be closed during a tornado warning, and two air-operated isolation dampers. These air intakes and exhaust structures are protected from tornado-generated and wind-generated missiles, rain, snow, ice, and trash. The exhaust vents from emergency DG areas are located at least 76.2 m (250 ft) from the nearest control room air intake to preclude diesel exhaust fumes from entering the MCRACS. The plant vent stack is located more than 61 m (200 ft) from the control room air intakes, as shown in CESSAR-DC Figure 3.8-5.

The MCRACS is a safety-related system consisting of an air conditioning system and an emergency circulation system. The air conditioning system consists of two redundant air conditioning units. The emergency circulation system consists of two redundant control room filtration units. Each system is fully redundant up to a common inlet header. The control room air conditioning unit has a prefilter, heating and cooling elements, and a fan. Each divisional control room filtration unit is an engineered safety features (ESFs) filtration unit which

consists of a moisture separator, prefilter, electric pre-heater, absolute high-efficiency particulate air (HEPA) filter, carbon adsorber, post-filter (HEPA), ducts and valves, and a fan as shown in CESSAR-DC Figures 9.4-1 and 9.4-2. Each divisional control room filtration unit conforms to the guidance of SRP Table 6.5.1-1 and RG 1.52, "Design Testing and Maintenance for Post Accident Engineered Safety Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," as stated in CESSAR-DC Tables 9.4-3A and 9.4-5. Dampers are up-stream and downstream of each ESF filtration unit and two air-operated, fail-closed dampers are in the emergency circulation system bypass ducts. Each of the redundant systems is powered from independent Class 1E, diesel-backed power sources, and cooling water for the AHU is supplied from the essential CWS. System components are accessible for periodic inspection. The non-safety-related TSCACS filter unit complies with the guidelines of RG 1.140, "Design, Maintenance and Testing Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," in entirety as the normal ventilation system. The MCRACS charcoal tray and screen is of all-welded construction to preclude the potential loss of charcoal from adsorber cells, as discussed in NRC IE Bulletin 80-03, Loss of Charcoal from Absorber Cells, February 6, 1980. All ducts work outside the CREZ of CCVS including the filtration units is either leaktight or is of welded construction. Ductwork will be periodically examined visually and pressure tested to maintain such positive pressure with respect to the adjacent areas that any unfiltered inleakages inside CREZ shall be less than the maximum allowable for the associated design. The system is designed to maintain the unfiltered leakage infiltration rate during pressurized operation of less than 0.005 m³/second (10 cfm). No steam piping exists adjacent to CREZ air intakes or inside CREZ, and no other HVAC system ducts, other than MCR air conditioning system ducts, pass through the CREZ.

During normal operation, the inlet air is continuously monitored for radiation, toxic gas, and smoke and is mixed with return air from the control room. The control room boundary pressurization system is tested every 18 months to verify that the makeup air required to maintain a positive minimum 3.2-mm (1/8-in.) water gauge pressure inside the control room boundary with respect to the adjacent areas does not exceed 10 percent of the design value. Pressure in the control room is maintained slightly positive relative to the surrounding areas and the outdoors at all times. The system design maintains the control room and other support areas between 23 °C and 26 °C (73 °F and 78 °F) and relative humidity between 20 and 60 percent; the battery room between 15.5 °C and 32.2 °C (60 °F and 90 °F); the mechanical equipment room is at

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40 °C (104 °F), and the remaining areas are at 29.4 °C (85 °F). The provisions of the minimum instrumentation and controls for the control room filtration units are listed in CESSAR-DC Table 9.4-3A. The provisions of the other instrumentation and controls monitor locally, and/or remotely: system temperatures, filter pressure drops, damper positions, chilled water flow rates, fan air flow rates and operating status, and high radioactivity and toxic gas at each outside air intake. The (1) system description, (2) design parameters, and (3) flow diagram are given in CESSAR-DC (1) Section 9.4.1, (2) Tables 9.4-1, 9.4-3, 9.4-3A, and 3.2-1, and (3) Figure 9.4-2, respectively.

The balance of control-complex air-conditioning systems serves the safety-related and non-safety-related areas. The safety-related areas comprise safety-related electrical rooms, vital instrument and equipment rooms, battery rooms, and the remote-shutdown room. These are served by individual redundant AHUs, each with roughing filters, safety-related chilled water cooling coils, and fans. The non-safety-related areas comprise: non-safety-related electrical rooms, battery rooms, operations and technical support centers, computer room, shift assembly and offices, radiation access control and central alarm station security group areas, personnel decontamination rooms, and a break room. These are served by individual air conditioning units, each with a roughing filter, non-safety-related chilled water cooling coils, and a fan. The safety-related and non-safety-related battery rooms have hydrogen detection devices to monitor hydrogen concentration. The exhaust fans in the battery room are designed to maintain concentrations of hydrogen gas below 2 percent, and their outlet ducts are located near ceiling. The redundant safety-related electrical, battery, and vital instrument and equipment room air conditioning systems are safety related and have fans to exhaust smoke vented on the roof of the control building. Safety-related systems receive cooling water from the safety-related CWS and are served by independent Class 1E, diesel-backed power sources. System components are accessible for periodic inspection.

The filtration unit for the emergency circulation system starts automatically if high radiation is detected at an air intake vent or if a SIAS is received. It filters the combination of the outside air and all of the return air and delivers the filtered air to the inlet of the main air conditioning unit which maintains the proper environmental conditions in the control room. If high radiation is detected at both inlet vents, the automatic selection logic compares the radiation levels at each inlet vent and closes the dampers in whichever inlet vent has the higher reading. Therefore, outside air to pressurize the control room comes automatically through the comparatively less-contaminated inlet. The pressurization and emergency filtration mode can also be actuated manually in the control room.

If the designated filtration unit fails to start, the redundant filtration unit starts automatically. At the same time, the component control logic isolates whichever control room intake has the greater radiation level, and blocks the isolation of whichever control room intake has the lesser radiation level.

On detection of toxic gases at the outside air inlet, the control room is automatically isolated from the outside environment. In the isolation mode of the system operation, the AHU maintains the control room at the proper environmental conditions and circulates the air in the room.

Redundant safety-related components of the MCRACS are physically separated and are protected from internally generated missiles, pipe breaks, and water spray. All safety-related components are seismic Class I, and function following a SSE. The system design maintains its function with the loss of any single active component.

In the event of a fire, the supply fan may be deactivated manually, if required, and smoke removal is then manually initiated from the control room or the remote-shutdown panel room by a combination of smoke exhaust fan, outside makeup air, and associated ductwork and control dampers. The fire dampers with fusible links in HVAC ductwork close under air flow conditions.

By virtue of this design, including location inside a seismic Category I flood-protected and tornado-missile-protected building, and classification of safety-related components as seismic Category I as shown in CESSAR-DC Table 3.2-1, the system complies with the requirements of RG 1.29, Position C.1, with respect to the system's safety-related portion and, thus, complies with GDC 2 as it relates to protection of the system against natural phenomena.

By virtue of the design for maintenance of environmental conditions and consideration of dynamic effects, the system complies with the requirements of GDC 4 as it relates to protection of the safety-related portions of the system against internally generated missiles and piping failures.

By virtue of use of redundant toxic-gas monitors, an alarm in the control room, and complying with the single-failure criterion for detection and isolation, the system complies with the requirements of GDC 19 regarding toxic gases. The toxic gases to be monitored will be determined by a COL applicant referencing the certified System 80+ design. Therefore, the specifics relating to compliance with GDC 19 as it pertains to protection of the control room against intrusion of toxic gases into the control room, are discussed in Section 6.4 of this report. As stated in that section, the compliance with GDC 19 in this regard is

demonstrated by complying with the guidance of RG 1.78 and 1.95 which, in turn, is within the scope of the referencing applicant since guidance is site dependent. The compliance with GDC 19 as it relates to radiation dose limits for the control room operator, is also discussed in Section 6.4 of this report.

Though ABB-CE indicated that the ESF-grade filter train is designed in accordance with the RG 1.52 guidelines, it was not clear if the filter train includes the minimum instrumentation requirements listed in SRP Table 6.5.1-1. This was designated as DSER Open Item 9.4.1-1. By CESSAR-DC amendments, ABB-CE has submitted CESSAR-DC Tables 9.4-3A, "Minimum Instrumentation, Readout, Recording and Alarm Provisions for ESF Atmosphere Cleanup Systems," and 9.4-5 "Design Comparison to Regulatory Positions of RG 1.52," which applies to the control complex ventilation system. The staff finds that the ESF-grade filter train design conforms to guidance in the RG and SRP Section 6.4. On this basis, DSER Open Item 9.4.1-1 is resolved.

Additionally, ABB-CE had not submitted information on the carbon adsorber efficiency for radioiodine removal. This was designated as DSER Open Item 9.4.1-2. By CESSAR-DC amendments, ABB-CE has submitted CESSAR-DC Table 9.4-3, "Input for Release Analysis Filter Efficiencies," showing the creditable carbon adsorber efficiency of 95 percent for the control room dose analysis and the activated carbon depth of the carbon filter as 51 mm (2 in.).

On this basis, DSER Open Item 9.4.1-2 is resolved. For these reasons, the system design complies with GDC 60 as it relates to the system's capability to control radioactivity.

On the basis of the preceding discussion, the staff concludes that the control-complex ventilation system complies with the applicable GDC 2, 4, 19, and 60 referenced in SRP Section 9.4.1 and consequently with the subject SRP acceptance criteria and is, therefore, acceptable.

9.4.2 Fuel Building Ventilation System

The staff reviewed the fuel building ventilation system (FBVS) in accordance with SRP Section 9.4.2 (NUREG-0800). The system is designed to mitigate the consequences of a postulated fuel-handling accident; maintain environmental conditions appropriate for equipment operation, perform maintenance and testing; and allow for personnel access and maintain building pressure below atmospheric to minimize out-leakage.

The FBVS is a once-through design which draws outdoor air through a damper and supply-air-handling unit, supplies

the air to building spaces, and exhausts the air to the outdoors through an exhaust fan. A bypass circuit of the exhaust system contains a filtration unit. The inlet supply AHU consists of a prefilter, cooling coil and electric heating coil, and one 100-percent capacity fan. The cooling coils are served by the NCWS. This portion of the system is not safety related nor is it serviced from the Class 1E power supply. The system is designed to maintain temperature between 4.4 °C and 40 °C (40 °F and 104 °F). The inlet-air vent is protected against wind and tornado missiles by missile shields above and in front of the opening. The (1) system description, (2) design parameters, and (3) flow diagram are given in CESSAR-DC (1) Section 9.4.2, (2) Tables 9.4-1, 9.4-3, 9.4-3A, 9.4-5, and 11.3-2, and (3) Figure 9.4-3, respectively. The system conforms to RG 1.52 for the particulate (HEPA) filtration credited during the fuel-handling accident to comply with 10 CFR Part 100 limits and particulate (HEPA) and elemental and organic iodine (carbon adsorber) filtration during normal operation as identified in CESSAR-DC Tables 9.4-3 and 9.4-5 and 11.3-2. SSCs are classified in CESSAR-DC Table 3.2-1 for the FBVS. The safety-related equipment, fans, dampers, coils and ductwork are designed and tested in accordance with ASME/ANSI N509 AG-1, "Nuclear Power Plant Air-Cleaning Units and Components"; N510, "Testing of Nuclear Air Treatment Systems"; codes; and standards. The radiological consequences resulting from gaseous effluent during normal plant operation, including anticipated operational occurrences, are discussed in Chapter 11 of this report.

The exhaust portion of the system is safety related (engineered safety features system), comprising two redundant 100-percent trains of fans and filtration units. During normal operation, air is released to the atmosphere through an exhaust fan and two control dampers. ABB-CE, in response to the staff's RAI Q410.117, stated that the single-bypass damper for the filtration system will be administratively locked closed and the system will be in operation whenever irradiated-fuel-handling operations above or in the fuel pool are in progress. This response was not acceptable since the single-failure criterion for these components must be complied with to prevent inadvertent release of radioactive contaminants to the environment. This was designated as DSER Open Item 9.4.2-1. In CESSAR-DC Section 9.4.2.2, ABB-CE states that the normal mode of operation does not require any filtration, and the bypass dampers are open for both the filtration trains. Upon receipt of a high-radiation signal, the system realigns the designated filtration train automatically to the filtration mode, to comply with 10 CFR Part 20 and 10 CFR Part 50 (Appendix I) requirements, by opening filtration unit inlet and outlet dampers and closing bypass dampers. Switchover between trains is

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accomplished manually. Before any fuel-handling operations, the system is manually realigned to the filtration mode and the bypass dampers are administratively locked closed. In this mode both filtration trains are aligned to process the effluent discharge before releasing through the monitored plant unit vent. The FBVS has two redundant 100-percent-capacity filtration trains which meet the single-failure criterion and fan- and motor-operated dampers in each train are powered from a separate train of the emergency Class 1E standby power in the event of any single active failure. The planned administrative isolation of the bypass dampers is not considered as an active function, and based on this, a single bypass damper in each train would continue to meet the single-failure criterion design for the exhaust side of the FBVS. On this basis, DSER Open Item 9.4.2-1 is resolved.

In CESSAR-DC Table 9.4-3, "Input for Release Analysis Filter Efficiencies," ABB-CE shows the creditable HEPA efficiency of 99 percent for the fuel-handling accident analysis. In CESSAR-DC Section 9.4.2.1, ABB-CE states that the dose analysis to support 10 CFR Part 100 limits following a fuel-handling accident only takes credit for the HEPA filter, and takes no credit for the charcoal adsorber.

The staff concluded in Section 15.A.11, of this report that with respect to the radiological consequences of potential fuel-handling accidents, credit is given for the removal of particulate iodines only. Therefore, charcoal adsorbers need not be credited in the FBVS.

A non-safety-related radiation monitor is located in the exhaust ductwork, upstream of the filter train inlet, which automatically directs the air through a filtration unit on detection of radioactivity in the duct. There is only one radiation detector, which is consistent with the guidance of RG 1.97, "Instrumentation for LWR Nuclear Power Plants To Assess Plant and Environs Conditions During and Following Accident." The redundant filtration unit comprises a moisture eliminator, prefilter, electric preheater, absolute (HEPA) filter, carbon adsorber (activated carbon depth of the carbon filter is 51 mm (2 in.)), post-filter (HEPA), ducts and valves, and a fan, as shown in CESSAR-DC Figures 9.4-1 and 9.4-3. Each division of the air-exhaust portion of the system has the capability to maintain the fuel-handling and fuel-storage areas at a negative pressure with respect to the atmosphere.

In the event of a fire, the exhaust and supply fans can be used for smoke removal. The fire dampers with fusible links in HVAC ductwork close under air flow conditions.

As described in CESSAR-DC Section 9.4.2.1 and Table 3.2-1, the system is located completely within a seismic Category I structure, and all safety-related components

(exhaust system and associated ductwork and filter train and fans) are seismic Category I, Safety Class 3, and the quality assurance requirements of 10 CFR Part 50 (Appendix B) are applicable. The flood protection, protection against internally and externally generated missiles, and high- and moderate-energy pipe breaks are evaluated in Sections 3.4.1, 3.5.1.1, 3.5.2, and 3.6.1 of this report. The equipment design and testing conform to the requirements of ANSI/ASME Standards AG-1, N509, and N510, and the equipment is available following an LOOP. Fan operating status and air flow rate, damper position, air temperatures, filter pressure drop, and chilled-water temperatures are monitored, and indicate either locally or in the control room. Failure of non-safety-related components does not compromise function of safety-related components.

ABB-CE committed to incorporate in CESSAR-DC Section 9.4.2 and Tables 8.3.1-2 and 8.3.1-3 the changes made in response to staff RAI Q410.1-17. This was designated as DSER Confirmatory Item 9.4.2-1. By CESSAR-DC amendments, ABB-CE stated that CESSAR-DC Tables 8.3.1-2 and 8.1.3-3 are revised to list the FBVS fans and filtration train electric heaters. There is a HEPA filter in each filtration train, downstream of the carbon adsorber. The FBVS safety-related equipment is tested in accordance with the ANSI/ASME Standards N509, N510, and AG-1. An in-service program will be implemented in accordance with 10 CFR Part 50 (Appendix B) and ASME Code Section XI, and a single radiation detector is provided for the entire exhaust portion of the FBVS per guidance of RG 1.97, Table 1. The single bypass damper in the exhaust portion of the FBVS is discussed in the resolution of DSER Open Item 9.4.2-1 above. On this basis, DSER Confirmatory Item 9.4.2-1 is resolved.

ABB-CE has reclassified the fuel handling area from seismic Category I to non-seismic in its response to staff RAI Q210.1. The ventilation system is located inside the fuel handling area (which is now part of the nuclear annex). ABB-CE was required to evaluate, and provide justifications concerning, the system protection against floods, hurricanes, tornados, and internally and externally generated missiles to conform with requirements of GDC 2. This was designated as DSER Open Item 9.4.2-2. By CESSAR-DC amendments, ABB-CE stated that the fuel building (handling area) is part of the nuclear annex structure which is classified as seismic Category I (Safety Class 3), as indicated in CESSAR-DC Section 9.4.2.1 and in CESSAR-DC Table 3.2-1. All safety-related FBVS components are fully protected from floods and tornado-missile damage, internal missiles, pipe breaks and whip, jet impingement and interaction of non-seismic systems in the vicinity. On this basis, DSER Open Item 9.4.2-2 is resolved.

In complying with the requirements of RGs 1.52 and 1.140 for controlling the release of radioactivity, the system also complies with the requirements of GDC 60 as related to the system capability to suitably control the release of gaseous radioactive effluents to the environment. The filtration units are designed in accordance with Position C.4 of RG 1.13, and consequently, the system complies with the requirements of GDC 61 as related to the system capability to provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment.

The staff concludes that the system conforms with applicable GDC referenced in SRP Section 9.4.2 and consequently, with the subject SRP acceptance criteria and is, therefore, acceptable.

9.4.3 Radwaste Building Ventilation System

The staff reviewed the radwaste building ventilation system (RWBVS) in accordance with SRP Section 9.4.3 (NUREG-0800). The RWBVS is a non-safety-related system and operates only during normal plant conditions. The RWBVS is located within the radwaste building except for the portion that connects with the unit vent at the juncture of the shield building and the nuclear annex. The radwaste building is maintained at a slight negative pressure with respect to the environment to ensure that all potentially radioactive releases are monitored before discharge. The system is a once-through cycle type.

The RWBVS comprises two 50-percent-supply AHUs, and cooling coils to supply normal ventilation and to control building temperature. The RWB ventilation exhaust system comprises two 50-percent particulate exhaust filter units each with moisture eliminator, prefilter, electric preheater, absolute (HEPA) filter, non-credited carbon adsorber, post (HEPA) filter, ducts and valves, and a fan. The system conforms to RG 1.140 for the filtration unit during normal operation as described in CESSAR-DC Table 9.4-6. The carbon filter medium conforms to nuclear grade as defined by the Institute for Nuclear Science. The radiological consequences resulting from gaseous effluent during normal plant operation including anticipated operational occurrences are discussed in Chapter 11 of this report.

The particulate and iodine radiation detectors which sample the air in ductwork, serve potentially occupied areas in which the potential for the release of radiation exists. These detectors also sample air in the exhaust duct header upstream of the filter units. Radioactivity above allowable limits indicates and alarms in the control room and locally. Upon detection of radioactivity above the allowable limit from the air exhaust, the bypass dampers will be manually

closed and the filter units' inlet and outlet dampers manually opened to allow the air exhaust filtration. The filtration exhaust fans discharge to the plant vent.

The system is designed to maintain temperature between 4.4 °C and 37.8 °C (40 °F and 100 °F). The system (1) description, (2) components, (3) design parameters, and (4) flow diagram are given in CESSAR-DC (1) Section 9.4.3, (2) Tables 9.4-1, and 9.4-3, (3) Section 11.3-2, and (4) Figure 9.4-9. SSCs are classified in CESSAR-DC Table 3.2-1 for the RWBVS. The safety-related equipment, fans, dampers, coils and ductwork are designed and tested in accordance with ASME/ANSI N509, N510, and AG-1 codes and standards.

The SSCs are non-nuclear safety but the structure housing and RWBVS (radwaste building) is designated as seismic Category II and is designed and constructed so that the SSE will not cause any failure in a manner that would adversely affect other safety systems as started in CESSAR-DC Section 3.2.1 and Table 3.2-1. Therefore, the system complies with GDC-2, as related to the system being capable of withstanding the effects of earthquakes.

In order to comply with GDC 60, "Control of Release of Radioactive Materials to the Environment," the system needs to conform to RG 1.140. Therefore, the RWB ventilation exhaust system HEPA filters should conform to RG 1.140, Positions C.1 and C.2. This was designated as DSER Open Item 9.4.3-1. In CESSAR-DC Section 9.4.3.1.1.H, ABB-CE states that the RWBVS conforms to the guidance of RG 1.140. On this basis, DSER Open Item 9.4.3-1 is resolved.

As revised in response to the staff's RAI Q210-1, ABB-CE states (in CESSAR-DC Table 3.2-1) the following (1) radwaste building ventilation system components are non-nuclear safety class, non-seismic, and the quality assurance requirements of 10 CFR Part 50 (Appendix B) are not applicable; and (2) the radwaste facility structure is seismic Category II as discussed in Chapter 3 of this report, and Appendix B quality assurance requirements are applicable. The staff evaluates flood protection, protection against internally and externally generated missiles, and protection against high- and moderate-energy pipe breaks in Sections 3.4.1, 3.5.1.1, 3.5.2, and 3.6.1 of this report.

In CESSAR-DC Section 9.4.3, ABB-CE describes the system but does not include design parameters and flow diagrams for system components. In the DSER, the staff stated that ABB-CE should submit this information so the staff can complete its review of the system. This was designated as DSER Open Item 9.4.3-2. ABB-CE submitted RWBVS flow diagrams for system components in CESSAR-DC Figure 9.4-9 and states in Section 9.4.3.2.1

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that the RWBV supply system consists of two 50-percent-capacity supply fans and the exhaust system consists of two 50-percent-capacity particulate filtration exhaust units. The components, as shown in CESSAR-DC Figures 9.4-1 and 9.4-9, include a non-credited carbon adsorber and exhaust fans. The staff concludes in Section 15.A.11 of this report that with respect to the radiological consequences of all potential accidents, ABB-CE gives credit for the removal of particulate iodines only. Therefore, charcoal adsorbers need not be credited in the RWBVS.

Air-flow rates of fans, operating status of fans, temperatures and flow rates of chilled water, damper positions/alignment, air-flow rates of supply and exhaust units, and air temperatures of supply ventilation units are monitored and indicate in the control room. The pressure drop across the supply filters and exhaust filtration trains is monitored and indicates locally and at the radwaste control panel.

The system is not safety related, performs no safety-related function for safe-shutdown or postaccident operation, and failure of the system does not affect the function of other safety-related equipment. Thus, the staff concludes that the RWBVS complies with the criteria GDC 2 and 60 of SRP Section 9.4.3 and is, therefore, acceptable.

9.4.4 Diesel Building Ventilation System

The staff reviewed the diesel building ventilation system (DBVS) in accordance with SRP Section 9.4.5 (NUREG-0800). The design has two redundant EDGs located in separate areas inside the nuclear annex on opposite sides of the reactor building. Each EDG area is served by a ventilation system designed to maintain acceptable environmental conditions for operation, testing, and maintenance of the equipment, and to allow for personnel access.

The DBVS is designed for once-through flow using inlet and exhaust fans, filters, and dampers. The system is designed to maintain temperature between a minimum of 4.4 °C (40 °F) and a maximum of 49 °C (120 °F) when the DG is not operating and between a minimum of 4.4 °C (40 °F) and a maximum of 50 °C (122 °F) when the DG is operating. Electric heaters, activated on low temperature, maintain temperature above freezing and fans are automatically activated to control elevated temperature. Air intake structures and exhaust vents are protected against the effects of natural phenomena and missiles.

Each division of the non-safety-related supply portion of the system consists of one 100-percent-supply fan equipped with a damper and prefilter. Air is exhausted to the outdoors through each division of the safety-related exhaust portion of the system which consists of two 50-percent-

supply fans. Each fan has a two-speed motor and a separate exhaust vent. The system is powered from a Class 1E source and is connected to Class 1E emergency standby power. The system is located in a seismic Category I building and all safety-related components are rated seismic Category I. In response to the staff's RAI Q210.1, ABB-CE revised CESSAR-DC Table 3.2-1 to identify the system space heaters and supply fan and their associated ductwork and dampers as Safety Class 3, seismic Category I; therefore, the quality assurance requirements of 10 CFR Part 50 (Appendix B) are applicable. The remaining components are designed as non-seismic safety class. The (1) system description, (2) design parameters, and (3) flow diagram are given in CESSAR-DC (1) Section 9.4.4, (2) Tables 9.4-1, 9.4-3, and 11.3-2, and (3) Figure 9.4-9, respectively. SSCs are classified in CESSAR-DC Table 3.2-1 for the DBVS.

In CESSAR-DC Section 9.4.4, ABB-CE describes the system; however, the component design parameters and flow diagram and the piping and instrumentation diagram were not submitted. This was designated as DSER Open Item 9.4.4-1. ABB-CE subsequently submitted the DBVS flow diagram in CESSAR-DC Figure 9.4-7 and the component design parameters in CESSAR-DC Table 9.4-1. On this basis, DSER Open Item 9.4.4-1 is resolved.

Additionally, ABB-CE committed to incorporate its response to staff RAI Q410.118 concerning the fresh-air intake vents, the seismic category of the building, space heaters, and system flow diagram. This was designated as DSER Confirmatory Item 9.4.4-1. Subsequently, ABB-CE stated that the location of the normal and emergency ventilation intakes is shown in Figure 1.2-6, the DG areas housing the system are designed to seismic Category I requirements, the space heaters are designed to maintain a minimum temperature of 4.4 °C (40 °F) as stated in CESSAR-DC Section 9.4.4.3, and the system flow diagram appears in Figure 9.4-7. The associated DG is started administratively to provide a substitute safety-related source of heat if the space heater fails to maintain the minimum design temperature. On this basis, DSER Confirmatory Item 9.4.4-1 is resolved.

No high-energy or moderate-energy piping is located near the ventilation equipment, and the inlet and exhaust dampers protect against external missiles. The fan indications and alarms, damper positions/alignment, and space temperature indications and alarms are located in the control room. The system design allows for periodic inspection. All safety-related components in the DBVS are designed to permit in-service inspection.

By virtue of location in seismic Category I areas, the system and its components are protected from natural

phenomena. All safety-related components are classified as seismic Category I. Also, failure of non-safety-related equipment will have no impact on the system. Therefore, the design complies with the requirements of GDC 2 as related to the system being capable of withstanding effects of earthquakes.

The system trains are located in a separate building above the DG, and system penetrations are protected from external missiles. Also, no high-energy or moderate-energy piping is located in the vicinity of the system. The design of safety-related components is compatible with environmental conditions and protects against dynamic effects; therefore, the design complies with the requirements of GDC 4 with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions.

Intake air is filtered for dust and other particulates, but the locations (elevations above ground level) of the intake vents were not specified for determining conformance with GDC 17, as it relates to ensuring proper functioning of the safety-related electric power system with respect to protection from the effects of dust and particulate materials. This was designated as DSER Open Item 9.4.4-2. Subsequently, in CESSAR-DC Section 9.4.4.3, ABB-CE states that the air intake is located 6.1 m (20 ft) above grade, as shown in CESSAR-DC Figure 1.2-6, to minimize intake of dust and particulate materials into the DG areas, and all exposed internal areas are painted to aid in dust control meeting the guidelines of Item 2, Subsection A, of NUREG-0600. Therefore, the system meets the pertinent requirement of GDC 17 for protection of the essential electrical components against dust accumulation and particulate material. On this basis, DSER Open Item 9.4.4-2 is resolved.

The system is physically separated from potentially contaminated areas and is not an internal source of radioactive materials. Therefore, the system conformance against the requirements of GDC 60 is not applicable.

The staff concludes that the DBVS complies with the applicable GDC 2, 4, and 17 referenced in SRP Section 9.4.5 and is, therefore, acceptable.

9.4.5 Subsphere Building Ventilation System

The staff reviewed the subsphere building ventilation system (SBVS) in accordance with SRP Section 9.4.5 (NUREG-0800). The SBVS is credited to limit the offsite and control room dose following a LOCA or DBA within the guidelines of 10 CFR Part 50 and SRP Section 6.4.

The radiological consequences resulting from gaseous effluent during normal plant operation (including anticipated operational occurrences) are discussed in Chapter 11 of this report; the radiological aspects are discussed in Chapter 15. The nuclear annex, fuel building, and reactor building subsphere are maintained under negative pressure with respect to the atmosphere.

The system consists of two supply and exhaust air divisions and serves all areas of the subsphere. The system is supplemented by the individual safety-related cooling units for the mechanical equipment room. Each of the two divisions serves one of the two plant divisions, providing redundancy of safety-related equipment. The supply and exhaust fans are electrically interlocked so that each SBVS division maintains its division of the subsphere building at a negative pressure relative to the atmosphere. A non-safety-related radiation detector monitors radioactivity of the area exhaust air in each divisional exhaust duct upstream of the exhaust filter train. The exhaust air is continuously processed through the divisional exhaust filter train and monitored before filtration and release through the unit vent. Additional gaseous radioactivity monitoring is provided in the unit vent. The (1) system description, (2) design parameters, and (3) flow diagram are given in CESSAR-DC (1) Section 9.4.5; (2) Tables 9.4-1, 9.4-3, 9.4-3A, and 11.3-2; and (3) Figures 9.4-4 and 9.4-5, respectively. The system conforms to RG 1.52 for the particulate (HEPA) filtration credited during the LOCA or DBA to comply with 10 CFR Part 100 limits and particulate (HEPA) and elemental and organic iodine (carbon adsorber) filtration during normal operation as stated in CESSAR-DC Tables 9.4-3 and 9.4-5 and 11.3-2. The classification of SSCs is provided in CESSAR-DC Table 3.2-1 for the SBVS. The safety-related equipment, fans, dampers, coils, and ductwork will be designed and tested in accordance with ASME/ANSI N509, N510, and AG-1 codes and standards.

In the DSER, the staff stated that CESSAR-DC Table 3.2-1 should appropriately identify the system, system components, and their locations with respect to safety class, seismic category, and quality assurance requirements designations. This was designated as DSER Open Item 9.4.5-1. Subsequently, ABB-CE submitted the requested information identifying the exhaust system as seismic Category I, Safety Class 3, and Quality Class 1, and the supply system as seismic Category II, non-nuclear safety class and Quality Class 2, except for the heating and cooling coils which are non-seismic, non-nuclear safety class for the Quality Class 3. On this basis, DSER Open Item 9.4.5-1 is resolved.

Outdoor air is drawn into the non-safety-related ventilation system serving each division through one 100-percent-

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capacity air-handling unit consisting of a prefilter, cooling/heating coils and two 100-percent-supply fans. The fresh air intake structures are located in the control areas duct shaft and are protected against high winds, rain, snow, and ice. The supply fans and conditioning unit are not safety-related units. Supply air is distributed to equipment rooms and access areas in the subsphere building and is exhausted from the building through a filtration unit by two 100-percent-capacity exhaust fans. The filtration unit and exhaust fans are safety-related equipment. The fans are powered from a Class 1E supply, backed up by the EDG.

Originally, ABB-CE did not submit information regarding the intake-air vent conformance with the GDC 17 requirement in GDC 17 as it relates to assuring proper functioning of the safety-related equipment, except for mentioning that the air is filtered. This was designated as DSER Open Item 9.4.5-2. In SRP Section 9.4.5, the staff offers guidance to ensure that adequate means is provided in the system design for control of airborne particulate material (dust) accumulation. The system arrangement should provide a minimum of 6.1 m (20 ft) from the bottom of the fresh air intakes to grade elevation.

Subsequently, ABB-CE submitted the missing information in CESSAR-DC Section 9.4.5.3, stating that the fresh air intakes are located at least 9.14 m (30 ft) above grade elevation (as shown in CESSAR-DC Figure 1.2-8) to minimize intake of dust into the building and that they have tornado dampers. On this basis, DSER Open Item 9.4.5-2 is resolved.

The divisional exhaust filtration unit consists of a moisture eliminator, prefilter, electric preheater, non credited carbon bed adsorber, and absolute and post filters (HEPA) upstream and downstream of the carbon adsorber as shown in CESSAR-DC Figures 9.4-1 and 9.4-5. A motor-operated damper on the downstream of exhaust fans is for tornado protection and for isolation when exhaust fans are off. In CESSAR-DC Table 9.4-3, "Input for Release Analysis Filter Efficiencies," ABB-CE shows the creditable HEPA efficiency of 99- percent for postaccident releases. ABB-CE states in CESSAR-DC Section 9.4.5.3 that the dose analysis, to support 10 CFR Part 100 limits following a LOCA or DBA, only takes credit for the HEPA filters in the filter train and takes no credit for the charcoal adsorbers.

The staff concludes that in CESSAR-DC Section 15.A.11, with respect to the radiological consequences of all potential accidents, ABB-CE give credit for the removal of particulate iodine only. Therefore, charcoal adsorbers need not be credited in the SBVS.

The staff concludes in Section 15.A.11 of this report, that with respect to the radiological consequences of all potential accidents, credit is given for the removal of particulate iodines only. Therefore, charcoal adsorbers need not be credited in the SBVS.

In addition to the air supply and filtration function, each divisional system has separate individual safety-related cooling units for each of the equipment rooms. The safety-related equipment comprises containment spray pumps and heat exchangers, safety injection system pumps and heat exchangers, shutdown cooling system pumps and heat exchangers, fuel pool heat exchangers, motor- and steam-driven emergency feedwater pumps, and penetration rooms. The safety-related cooling units recirculate air through prefilters, cooling coils serviced from the essential CWS, and fans. The AHUs for the safety-related equipment rooms AHUs are powered by a Class 1E source, backed up by the EDG. All cooling units are started automatically and remain operational throughout a LOCA. All safety-related system components are designed to permit in-service inspection. The cooling units for the safety-related equipment rooms are designed to maintain the temperature below 38 °C (100 °F). At least one train of safety-related equipment rooms is maintained below 38 °C (100 °F), assuming a single failure of an active component concurrent with an LOOP. Redundant components of the cooling systems for the safety-related equipment rooms are physically separated and protected from internally generated missiles. A pipe break in the same safety-related mechanical train is the only possible means of affecting the essential CWS. Therefore, when subjected to the effects of pipe breaks, the components are not required to operate because the served equipment is located in the same space as the cooling components. All safety-related equipment is rated seismic Category I, is located in a seismic Category I building, and will remain functional following a design basis earthquake. The components are protected from tornados, and intake and exhaust vents are protected from rain, snow, and ice. The ductwork from the building exit up to and including the isolation damper are qualified for the tornado differential pressure. Failure of the non-safety-related supply units will not affect the safety function of the safety-related units.

The minimum instrumentation for the ESF exhaust filtration trains is listed in CESSAR-DC Table 9.4-3A. Air flow rates of fans, operating status of fans, temperature and flow rate of chilled water, damper positions/alignment, and air temperatures of supply ventilation units are monitored and indicate in the control room. The pressure drop across the supply filters and exhaust filtration trains is monitored and indicates locally. The equipment is inspected periodically and the design allows for in-service inspection.

Because safety-related components are classified as seismic Category I and located within a seismic Category I structure, the ventilation system complies with the requirements of GDC 2 as related to the system being capable of withstanding the effects of earthquakes.

By virtue of design with respect to maintenance of environmental conditions and consideration of dynamics effects, the system complies with the requirements of GDC 4 with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment during normal, transient, and accident conditions.

As indicated above, the system is consistent with the requirements of RGs 1.140 and 1.52 for exhaust system cleanup and filtration and, therefore, complies with the requirements of GDC 60 as related to the system capability to suitably control the release of gaseous radioactive effluents to the environment.

ABB-CE committed to incorporate in the CESSAR-DC its response to staff RAI Q410.119 concerning the in-service testing, exhaust fan failure, fresh air intakes, and compliance with RGs 1.140 and 1.52. This was designated as DSER Confirmatory Item 9.4.5-1. Subsequently, ABB-CE stated that (1) all safety-related subsphere ventilation system components are designed to permit in-service inspection; (2) the failure of the non-safety-related supply fan has no effect on the exhaust fan since negative pressure is maintained inside the subsphere and the exhaust is filtered; (3) the fresh air-intakes are located at least 9.14 m (30 ft) above grade elevation and protected against adverse environmental conditions; (4) a HEPA filter is placed downstream of the carbon adsorbers, and filtration components in each filtration train are shown in accordance with CESSAR-DC Figure 9.4-1 to comply with RG 1.52; and (5) the system contains differential pressure alarms and indication in compliance with the guidance of RG 1.140, as referenced in CESSAR-DC Sections 9.4.5.1 and 9.4.5.3. On this basis, DSER Confirmatory Item 9.4.5-1 is resolved.

Based upon the above discussion, the staff concludes that the SBVS complies with GDC 2, 4, 17 and 60 referenced in SRP Section 9.4.5 and is, therefore, acceptable.

9.4.6 Containment Cooling and Ventilation System

The staff reviewed the containment cooling and ventilation system (CC&VS) in conformance with GDC 2, as related to the system being capable of withstanding the effects of earthquakes, referenced in SRP Section 9.4.5 (NUREG-0800). This system maintains suitable environmental

conditions inside the containment for normal operation, maintenance, and testing. The system is non-safety related except for high-and-low purge CIVs and penetration ductwork that isolate portions of the system inside the containment from portions of the system in the nuclear annex up to the HEPA filters. The low-purge and high-purge systems are designed to maintain the containment under slight negative pressure with respect to the atmosphere.

The CC&VS comprises (1) the recirculation cooling system, (2) the low-and high-purge supply and exhaust subsystems, (3) the containment air cleanup system, (4) the pressurizer compartment cooling redundant fans, (5) the reactor cavity compartment cooling redundant fans, and (6) the CEDM cooling system.

The recirculation cooling system consists of four 33-percent-capacity recirculation cooling units. The recirculation cooling units remove heat from the containment that is generated by the nuclear steam supply system support structures and RCS insulation heat loads (SSAR Tables 9.4-4 and 9.4-2), and maintain the served areas between 15.5 °C and 43.3 °C (60 °F and 110 °F). Heat energy is transferred to a non-essential leg of the chilled-water system through the chilled-water cooling coils located in the recirculation cooling units.

The low-purge subsystem relieves containment pressure during startup or shutdown. The IRWST purge supply and exhaust is normally closed and is opened only for personnel access. The high-purge system operates to reduce radiation levels before and during personnel access to the containment. The containment high-purge system (HEPA filtration only) mitigates the radiological consequences of a postulated fuel-handling accident inside the containment to conform with 10 CFR Part 100 requirements and is not used during power operation. The low-purge subsystem (HEPA filtration) mitigates the consequences of a control element ejection accident inside containment to conform with 10 CFR Part 100 requirements.

The containment air cleanup system consists of a prefilter, an absolute HEPA filter, a carbon adsorber, a post-HEPA filter, and a fan. It is designed to reduce containment airborne concentrations to approximately seven maximum permissible concentrations (MPCs) to permit personnel access, and it conforms to ANSI/ANS-56.6, "Pressurized Water Reactor Containment Ventilation Systems."

Fans in the reactor cavity and pressurizer compartments in conjunction with the recirculation cooling system, maintain temperature in these areas below 54.4 °C (130 °F).

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The CEDM cooling system consists of redundant cooling units. The CEDM cooling system maintains temperature in the served areas to 76.6 °C (170 °F).

In the event of fire, the exhaust and supply fans of the purge systems can be used for smoke removal. The (1) system description, (2) design parameters, and (3) flow diagram are given in CESSAR-DC (1) Section 9.4.6, (2) Tables 9.4-1, 9.4-3, 9.4-3A, and 11.3-2, and (3) Figure 9.4-6, respectively. The system conforms to RG 1.52 for the credited high-volume purge system HEPA filtration during a postulated fuel handling accident and low-volume purge system HEPA filtration during a control element ejection accident to comply with 10 CFR Part 100 limits. The system conforms to RG 1.52 for the particulate (HEPA) and elemental and organic iodine (carbon adsorber) filtration for the high-volume containment purge during normal operation. The system conforms to RG 1.140 for the particulate (HEPA) and elemental and organic iodine (carbon adsorber) filtration for the low-volume containment purge during normal operation as identified in CESSAR-DC Tables 9.4-5 and 9.4-6. SSCs are classified in CESSAR-DC Table 3.2-1 for the CCVS. High and low purge exhaust ductwork from the containment penetration to the HEPA filters, penetration ductwork and CIVs are seismic Category I safety Class 2 and Quality Class 1. The radiological consequences of gaseous effluent during normal plant operation including anticipated operational occurrences are discussed in Chapter 11 of this report. The safety-related equipment, fans, dampers, coils, and ductwork are designed and tested in accordance with RG 1.52 and ASME/ANSI N509, N510, and AG-1 codes and standards.

The air intakes are protected against tornado-generated external missiles as shown in CESSAR-DC Figure 1.2-10A. The motor-operated dampers downstream of the high- and low-purge exhaust systems are manually closed during a tornado warning and are used for isolation when exhaust fans are not in operation.

Outdoor air enters the low-purge system through an air conditioning unit and supply fans located in the nuclear annex. The air conditioning unit comprises a prefilter, an absolute (HEPA) filter, and a heating coil. Air from the fan passes through a fail-closed pneumatically operated isolation damper located in the annulus and an isolation damper located in the containment before it is distributed into the containment atmosphere. Exhaust air passes through two more fail-closed pneumatically operated isolation dampers, one each in the containment and annulus, and then through the exhaust filter train. The exhaust filter unit and fans are located in the nuclear annex. The air-exhaust filter unit comprises a prefilter, a heating coil, a non-credited carbon adsorber, and HEPA

filters upstream and downstream of the carbon adsorber, as shown in CESSAR-DC Figures 9.4-1 and 9.4-6. The location and sequence of equipment of the high-purge subsystem parallels that of the low-purge system with the following exceptions: (1) the high-volume purge system has two penetrations for each high-purge supply and exhaust train and (2) each penetration has two fail-closed pneumatically operated isolation dampers, one in the annulus and one in the containment.

In response to the staff's RAI Q410.120, ABB-CE stated that the containment high volume purge system is not an engineered safety features system. During a postulated fuel-handling accident inside containment, the charcoal filtration is credited with filtration of the release, but no credit is allowed for release reduction resulting from containment isolation or mixing in the containment atmosphere preceding the release. In the CESSAR-DC, ABB-CE committed to incorporate the response. This was designated as DSER Confirmatory Item 9.4.6-1. Subsequently, ABB-CE states in CESSAR-DC Sections 9.4.6.1 and 9.4.6.3 that the dose analysis demonstrating conformance with 10 CFR Part 100 limits following a fuel-handling accident and a control element ejection accident, only takes credit for the HEPA filtration. No credit is taken for the charcoal adsorbers in either of the containment exhaust paths.

The staff concludes in Section 15.A.11 of this report, that with respect to the radiological consequences of all potential accidents. ABB-CE gives credit for the removal of particulate iodines only. Therefore, charcoal adsorbers need not be credited in the CC&VS.

By CESSAR-DC amendments, ABB-CE revised CESSAR-DC Table 3.2-1 in response to the staff's RAI Q210.1 to show that the safety-related components (11 containment isolation dampers and the associated penetration ductwork from the containment penetration to the filter trains) are Safety Class 2 and seismic Category I, and that the quality assurance requirements of 10 CFR Part 50, (Appendix B) are applicable. The remaining components are not safety related. All 11 containment isolation dampers are normally closed, are designed to fail closed, and receive a containment isolation signal to close. Additionally, the high-volume purge system remains sealed closed during power operation. The containment purge exhaust system is isolated on high radiation signal or high relative humidity signals.

Air flow rates, area temperatures, and chilled water flows and temperatures for the containment recirculation cooling units and CEDM cooling unit fans, operating status of fans and alarms, damper positions/alignment, temperatures of various representative areas inside the containment, and

high vibration alarms for all fans located inside the containment are monitored and indicated in the control room. The pressure drop across all filters is monitored and indicated locally.

Because seismic Category I components are used for safety-related equipment and are located in a seismic Category I structure, the system complies with the requirements of GDC 2 as defined previously.

The review established that the system is not an ESF and except as noted above, the system is not credited in analyzing the consequences of DBA. Therefore, the requirements of GDC 4, 17, and 60 are not applicable to this system.

The staff concludes that the CC&VS complies with the applicable GDC referenced in SRP Section 9.4.5 and is, therefore, acceptable.

9.4.7 Turbine Building Ventilation System

The staff reviewed the turbine building ventilation system (TBVS) in accordance with SRP Section 9.4.4 (NUREG-0800) and industry standards. As identified in CESSAR-DC Table 3.2-1, the turbine building ventilation system components are non-nuclear safety class, non-seismic category, and the quality assurance requirements of 10 CFR Part 50 (Appendix B) do not apply. The turbine building is seismic Category II, as discussed in Chapter 3 of this report.

In the DSER, the staff stated that CESSAR-DC Section 9.4.7 described the system briefly, giving no design parameters for system components or flow diagrams. This was designated as DSER Open Item 9.4.7-1. Subsequently, ABB-CE stated that the system consists of fans, intake dampers, exhaust fans, and ductwork, and that recirculation fans create a uniform mixture. Also, radiation monitoring information is given in CESSAR-DC Section 11.5. The TBVS removes the heat dissipated by equipment, piping, lighting, and solar heat gains, and maintains the served areas between 4.4 °C and 43.3 °C (40 °F and 110 °F). The design outside temperature are based on the 5-percent exceedance air temperature values. The (1) system description and (2) layout drawings are given in (1) CESSAR-DC Section 9.4.7 and (2) Figures 1.2-18 and 1.2-19, respectively. SSCs for the TBVS are classified in CESSAR-DC Table 3.2-1. On this basis, DSER Open Item 9.4.7-1 is resolved.

The system instrumentation for manual and automatic operations and system verification is provided locally. The fan indications and alarms are also provided locally.

The review established that the system is not safety related and that failure of the system does not compromise the operation of safety-related systems. Therefore, the requirements of GDC 2, 4, 17, and 60 are not applicable to this system.

However, the staff finds that the system alternately conforms with applicable industry standards, to conclude acceptability of the non-safety system for the proper environment during normal operation, including: Fan ratings conform to the Air Moving and Conditioning Association (AMCA) Standards, Fan motors conform to applicable standard of the National Electrical Manufacturers Association (NEMA) and the Institute of Electrical and Electronic Engineers (IEEE), Ventilation ductwork conforms to applicable standards of the Sheet Metal and Air Conditioning Contractors National Association (SMACNA), Applicable components and controls conform to the requirements of IEEE, Underwriter's Laboratories (UL) and NEMA and Heating coil ratings conform to standards of the Air Conditioning and Refrigeration Institute (ARI).

The staff concludes that the TBVS complies with the acceptance criteria of the above industry standards and SRP Section 9.4.4 and is, therefore, acceptable.

9.4.8 Station Service Water Pump Structure System

In response to staff RAI Q410.121, ABB-CE stated that the station service water pump structure ventilation system (SSWPSVS) is dependent on site-specific considerations. ABB-CE committed to provide interface requirements for the SSWPSVS. This was designated as DSER Confirmatory Item 9.4.8-1. Subsequently, ABB-CE submitted a conceptual design and interface requirements in CESSAR-DC Section 9.4.8.1.2. On this basis, DSER Confirmatory Item 9.4.8-1 is resolved.

SSCs for the SSWPSVS are classified in CESSAR-DC Table 3.2-1. The conceptual engineered safety feature (ESF) system design includes two safety-related, 100-percent-supply fans with the associated dampers, ducts, and instrumentation to ensure that the system remains functional during a loss of offsite power and meets the single-failure criterion. The system instrumentation for manual and automatic operations and system verification is provided in the main control room and locally. The fan indications and alarms, and space temperature indication with high/low alarms, are provided in the main control room.

The SSWPSVS interface requirements, when met, will ensure that the system is designed in accordance with the requirements of GDC 2, as related to the system being

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capable of withstanding the effects of earthquakes (Acceptance is based on meeting the guidance of RG 1.29, position C.1 for safety-related portions and C.2 for non-safety-related portions), GDC 4, with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions, GDC 5, as related to shared system and components important to safety, GDC 17, as related to assuring proper functioning of the essential system (Acceptance is based on meeting the guidance of item 2 under subsection A and item 1 under subsection C of the section on "Recommendations" of NUREG-CR/0600 relating to the protection of essential electrical components from failure due to the accumulation of dust and particulate materials) and GDC 60, as related to the system capability to suitably control release of gaseous radioactive effluents to the environment Acceptance is based on meeting the guidance of RGs 1.52 and 1.140, as related to design testing, and maintenance criteria for atmosphere cleanup system, and normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants, Position C.2 and Positions C.1 and C.2, respectively).

The staff concludes that the ABB-CE SSWPSV conforms to the applicable GDC referenced in SRP Section 9.4.5, and subsequently the subject SRP acceptance criteria and the conceptual design for the system is therefore acceptable.

9.4.9 Nuclear Annex Ventilation System

The staff reviewed the nuclear annex ventilation system (NAVS) in accordance with SRP Section 9.4.5 (NUREG-0800). The safety-related cooling systems for the mechanical equipment room, serving the CCWS pump rooms and the safety-related CWS system pump and chiller rooms, are designed to seismic Category I requirements and remain functional following a design-basis earthquake (DBE). These cooling systems consist of two completely redundant, independent, full-capacity systems; they are able to withstand the effects of tornados, floods, hurricanes, and fires. In the event of a LOCA or DBA, the safety-related mechanical equipment room cooling and ventilation units operate at full capacity throughout the event and maintain space temperatures below 37.8 °C (100 °F). The design temperature range for the non-safety-related areas is 15.6 °C (60 °F) to 37.8 °C (100 °F).

The system comprises two general air supply and exhaust systems, each of which is supplemented by safety-related and non-safety-related recirculation cooling units to serve the safety-related areas (described above), and non-safety-related mechanical equipment areas. The non-safety-related areas contain the normal chillers and pumps rooms,

CVCS tanks and pump rooms, hot machine and tool rooms, the sample room, and instrument air rooms. The two systems independently service Division I and II areas. The general supply and exhaust system functions to maintain suitable environmental conditions throughout the nuclear annex. The (1) system description, (2) design parameters, and (3) flow diagram are given in CESSAR-DC (1) Section 9.4.9, (2) Tables 9.4-1 and 11.3-2, and (3) Figure 9.4-8, respectively. The system conforms to RG 1.52 for the HEPA filtration credited during the LOCA or DBA to comply with 10 CFR Part 100 limits. The system conforms to RG 1.140 for the filtration unit during normal operation as identified in CESSAR-DC Tables 9.4-5 and 9.4-6. SSCs are classified in CESSAR-DC Table 3.2-1 for the NAVS. The safety-related equipment, fans, dampers, coils and ductwork are designed and will be tested in accordance with RG 1.52 and ASME/ANSI N509, N510, and AG-1 codes and standards. The nuclear annex building is always maintained at a slight negative pressure with respect to the outside atmosphere to ensure that all potential radioactive releases are monitored before the air supply is discharged to the atmosphere. The radiological consequences resulting from gaseous effluent during normal plant operation including anticipated operational occurrences are discussed in Chapter 11 of this report.

Outdoor air is drawn into the system serving Division I areas through a 100-percent-supply unit by two 100-percent-supply fans. The supply unit has a prefilter, heating and cooling coils, and a bypass flow loop. The outside air flow to the inlet unit is controlled by three motor-operated dampers. Air is distributed throughout the nuclear annex and discharged through a 100-percent-filtration unit by two 100-percent-capacity exhaust fans. In the event of a fire, the exhaust and supply fans can be used for smoke removal. The filtration unit has a bypass and comprises a moisture eliminator, a prefilter, a heating coil, a non-credited carbon adsorber, and a HEPA filter upstream and downstream of the carbon adsorber, as shown in CESSAR-DC Figures 9.4-1 and 9.4-8. Flow through the filtration unit is controlled with three motor-operated dampers. The exhaust air is monitored by a radioactive gaseous detector located upstream of the filtration train. Upon detection of activity, the exhaust is processed through the divisional filtration train before discharge to the atmosphere through the plant vent where additional monitoring of the exhaust takes place. The exhaust damper is located upstream of the discharge to the unit vent in each filter train and is manually closed during a tornado warning.

Recirculation cooling units are in the CCWS pump rooms, and in the safety-related CWS pump and chiller rooms. The recirculation units include prefilters, cooling coils, and

fans. Cooling water is supplied by the safety-related CWS. Ventilation and cooling equipment serving Division II is identical to the system serving Division I, except the Division II consists of two 50-percent-capacity particulate filtration units and two 100-percent-capacity exhaust fans for each filtration unit. The systems are physically separated.

The safety-related recirculation units for the mechanical equipment rooms are fully redundant and remain functional with the loss of any single active component. Each train is powered from an independent, Class 1E power source. The safety-related recirculation units are inherently separated from non-safety-related units, and the safety function of the filtration units and fans is unaffected by failure of other system components. An LOOP will not affect the safety function of safety-related equipment.

The system is inspected periodically and the design allows for in-service inspection of safety-related components. Fan air-flow rates and operating status, damper position, room temperatures, and chilled water temperatures are monitored and indicated in the control room. The filtration unit pressure drop is monitored and indicated locally.

ABB-CE revised CESSAR-DC Table 3.2-1 in response to the staff's RAI Q210.1 to identify (1) the nuclear annex structure as seismic Category I; (2) safety-related recirculation units and room recirculating unit cooling coils as Safety Class 3, seismic Category I; and (3) the quality assurance requirements of 10 CFR Part 50 (Appendix B) as applicable. The remaining components are not safety related. All safety-related components are designed as seismic Category I equipment and are located in a seismic Category I structure. The safety-related components are operated from a Class 1E power source and function on LOOP. The safety-related components are protected from externally and internally generated missiles. The staff evaluated flood protection, protection against internally and externally generated missiles, and high- and moderate-energy pipe breaks in Sections 3.4.1, 3.5.1.1, 3.5.2, and 3.6.1 of this report. The filtration units provide the means to control leakage, and the cooling units are activated by room temperature.

Because safety-related components are classified as seismic Category I and are located within a seismic Category I structure, the ventilation system complies with the requirements of GDC 2 as previously discussed.

By virtue of design with respect to maintenance of environmental conditions and consideration of dynamic effects, the system complies with the requirements of GDC 4 as previously discussed.

In the DSER, the staff stated that ABB-CE had not submitted information regarding the conformance of the intake vents to GDC 17 requirements relating to ensuring proper functioning (protection against airborne particulate material) of the safety-related equipment, except for mentioning that the air is filtered. This was designated as DSER Open Item 9.4.9-1. Subsequently, in CESSAR-DC Section 9.4.9.3, ABB-CE states that the system fresh air intakes are located at least 9.14 m (30 ft) above grade elevation to minimize intake of dust into the building. In CESSAR-DC Figure 1.2-8, ABB-CE shows the divisional air intake locations. On this basis, DSER Open Item 9.4.9-1 is resolved.

As indicated in the previous material, the system is consistent with the requirements of RGs 1.140 and 1.52 for exhaust system cleanup and filtration and, therefore, complies with the requirements of GDC 60 as previously discussed.

ABB-CE committed to incorporate the response to the staff's RAI Q410.122 concerning the physical location and configuration of the system components, in-service inspection for safety-related components, and safety-related equipment function during LOOP in CESSAR-DC Section 9.4.9. This was designated as DSER Confirmatory Item 9.4.9-1. Subsequently, ABB-CE stated that (1) the heat loads and design parameters are presented in CESSAR-DC Table 9.4-1; (2) the physical location of major components is shown in CESSAR-DC Figure 1.2-8 and the system flow is diagrammed in CESSAR-DC Figure 9.4-8; (3) all safety-related components are designed to permit in-service inspection as stated in CESSAR-DC Section 9.4.9.4; (4) the nuclear annex structure is designed to seismic Category I standards; and (5) an LOOP will not affect the safety function of safety-related equipment. On this basis, DSER Confirmatory Item 9.4.9-1 is resolved.

Based on the above, the staff concludes that the NAVS complies with the applicable GDC 2, 4 and 60 referenced in SRP Section 9.4.5 and consequently with the SRP acceptance criteria and is, therefore, acceptable.

9.4.10 Component Cooling Water Heat Exchanger Structure(s) Ventilation Systems

The staff reviewed the CCW heat exchanger structure(s) ventilation systems (HXSVS) in accordance with SRP Section 9.4.5 (NUREG-0800) and industry standards. As identified in CESSAR-DC Table 3.2-1, the CCWHXSVS components are located completely within seismic Category I structures, and fans, dampers, and ductwork are protected from floods and tornado-missile damage and interaction with other non-seismic systems. The fans, dampers, ductwork, unit heaters, and supports are designed as

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seismic Category II, non-nuclear safety class, and the quality assurance requirements of 10 CFR Part 50 (Appendix B) do not apply. The system is not required to operate in order for the CCWS to perform its safety function. The CCW heat exchanger structure(s) which houses the system is seismic Category I, Safety Class 3, and the quality assurance requirements of 10 CFR Part 50 (Appendix B) apply.

Two CCWHXSVS are provided, one for each division of CCW. The two divisions are physically separated and there is no interaction between the systems. Each division consists of a fan, associated motor-operated intake and exhaust dampers, ductwork, supports, and instrumentation and controls. The (1) system (2) design parameters, and (3) flow diagram are described in (1) CESSAR-DC Section 9.4.10, (2) Table 9.4-1, and (3) Figure 9.4-10. SSCs are classified in CESSAR-DC Table 3.2-1 for the CCWHXSVS.

The system maintains the served areas above 4.4 °C (40 °F). The system fresh air intakes are located at least 6.1 m (20 ft) above grade elevation and away from plant discharges to minimize intake of dust and contaminants into the structures.

The CCWS fluid is monitored by radiation detectors and any radioactivity present is contained within the piping system; therefore, no provisions are made to contain the release of the radioactive materials in the CCWHXSVS.

The system instrumentation for manual and automatic operations and system verification is provided remotely or locally. The fan indications and alarms are provided in the control room. The space temperature indication and high/low alarms are also provided in the control room.

The review established that the system is not safety related and that failure of the system does not compromise the operation of safety-related systems. Therefore, the requirements of GDC 2, 4, 17, and 60 are not applicable to this system. However, the staff finds that the system alternately conforms with applicable industry standards to conclude acceptability of the non-safety system for the proper environment during normal operation, including: Fan ratings conform to the Air Moving and Conditioning Association (AMCA) Standards, fan motors conform to applicable standard of the National Electrical Manufacturers Association (NEMA) and the Institute of Electrical and Electronic Engineers (IEEE), ventilation ductwork conforms to applicable standards of the Sheet Metal and Air Conditioning Contractors National Association (SMACNA) and applicable components and controls conform to the requirements of IEEE, Underwriter's Laboratories (UL) and NEMA.

Based on the above, the CCWHXSVS complies with the acceptance criteria of the above industry standards and SRP Section 9.4.5 and is, therefore, acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire-Protection System

The Commission directed that special attention be given to measures for fire protection in addition to the staff's review of other aspects of the ABB-CE System 80+ design in accordance with the requirements for current operating plants. For example, the Commission concluded that the ABB-CE design must incorporate the resolution of significant fire protection issues raised through operating experience and through the External Events Program.

The NRC established fire protection requirements for nuclear power plants in GDC 3, 10 CFR 50.48, and Appendix R to 10 CFR Part 50. The Commission considered Section III.G, III.J and Appendix R to be of particular importance. In July 1981, NRC revised BTP APCSB 9.5-1 (SRP Section 9.5.1) to include these provisions from Appendix R.

The staff has also issued supplemental guidance on fire protection in documents such as Generic Letter (GL) 81-12 (45 FR 76602, November 19, 1981), dated February 20, 1981, and GL 86-10, dated April 24, 1986. GL 81-12 presents information on safe-shutdown methodology and GL 86-10 presents technical information on conformance with National Fire Protection Association codes and standards.

To minimize fire as a significant contributor to the likelihood of severe accidents for the ABB-CE design, the staff concluded that current NRC guidance must be enhanced. As stated in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990 (and reiterated in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs"). The proposed applicable regulation discussed below encompasses the enhanced NRC Guidance.

The NRC expects any new reactor design to propose fire protection systems based on the best technology available, not on the methods allowed for plants already operating or in the advance stages of design and construction. Therefore, the staff evaluated the fire protection system of the System 80+ design against the criteria of SRP Section 9.5.1 (BTP CMEB 9.5-1 Rev. 2), which meets the requirements of GDC 3, and against the following proposed applicable regulation for fire protection: The

standard design must comply with 10 CFR Part 50, Appendix R, Section III.G.1.a and ensure that safe shutdown can be achieved assuming all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. The design must ensure that smoke, hot gasses, or fire suppressant, will not migrate into other fire areas to the extent that could adversely affect safe shutdown capabilities including operator actions. The control room is excluded because an alternative shutdown capability is provided which is physically and electrically independent of the control room. In the reactor containment, redundant shutdown systems must be provided with fire protection to ensure, to the extent practical, that one shutdown division be free of fire damage. Because of unique design layout, other areas may be accepted on an individual basis.

9.5.1.1 General Evaluation of the Fire-Protection Program

ABB-CE generally follows the NRC's concept of defense in depth with regard to fire protection as described in the CESSAR-DC. The three steps of defense in depth and ABB-CE's implementation of these steps follow:

- To reduce the possibility of fire starting in the plant, ABB-CE uses fire-resistant and fire-retardant materials in its design to minimize and isolate fire hazards. ABB-CE uses low-voltage multiplexed circuits or fiber-optic circuits, or both, in its design to eliminate the need for cable spreading rooms. This substantially reduces the amount of combustible cable insulation and higher voltage ignition sources in the control room.
- To promptly detect and suppress fire, ABB-CE provides adequate automatic detection and a suitable mix of automatic and manual fire-suppression capability in the System 80+ design.
- To ensure that any fire that might occur does not prevent safe shutdown of the plant, even if fire detection and suppression efforts should fail, ABB-CE submitted a fire-protection program in its application.

The fire-protection program described in the CESSAR-DC is intended to protect safe-shutdown capability, prevent release of radioactive materials, minimize property damage, and protect personnel from injury as a result of fire.

In addition to the three aspects of defense in depth listed above, ABB-CE also considered such features of general plant arrangement as:

- access and egress routes
- equipment locations
- structural design features separating or isolating redundant safety-related systems
- floor drains
- ventilation
- construction materials

ABB-CE reflected the use of applicable NFPA codes and standards in its design and layout of the System 80+. In the DSER, the staff asked ABB-CE to identify deviations from the NFPA codes and standards and to describe in the Fire Hazards Assessment the deviations and measures taken to ensure that equivalent protection is provided. This was designated as DSER Open Item 9.5.1.1-1.

In a letter dated June 11, 1993 (LD-93-090), ABB-CE indicated that there are very few code deviations in the fire-protection design of the System 80+.

ABB-CE noted one deviation for which the NRC has regulatory authority. That deviation concerns the containment isolation valves (CIVs) for the fire-protection piping. ABB-CE stated that nuclear safety valves are not available that have been tested and approved or listed by a nationally recognized testing laboratory for fire-protection purposes. However, ABB-CE indicated that the nuclear safety valves will meet or exceed the construction requirements of listed and approved fire-protection valves. ABB-CE will utilize nuclear safety valves that meet or exceed the construction requirements for fire-protection valves. Although the nuclear safety valves are not listed by a nationally recognized testing laboratory, it is expected that these valves will function as required and will be monitored under the fire-protection valve-surveillance program.

ABB-CE's approach to use nuclear safety valves for limited applications for containment isolation in the fire-protection system piping, in lieu of approved or listed fire-protection valves, is adequate. However, final acceptability will be based on review and acceptance of the specific valves as part of the staff's review of the plant-specific fire hazards analysis as discussed in Section 9.5.1.6 in this chapter.

ABB-CE stated in a letter dated June 11, 1993, (LD-93-090), that although there are few deviations from NFPA codes and standards, other deviations may be identified during later detailed design after design certification. These deviations will be addressed in the fire hazards analysis to be submitted by the COL applicant. ABB-CE's approach to address deviations to NFPA codes and standards is acceptable. On this basis, DSER Open Item 9.5.1.1-1 is resolved.

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9.5.1.2 Specific Features of Protection

9.5.1.2.1 Protection of Safe-Shutdown Capability

9.5.1.2.1.1 Fire Outside the Containment and the Control Room

Outside of the containment, the System 80+ design plant configuration completely separates redundant shutdown Divisions I and II for all fire areas except for the control room and the remote shutdown panel (RSP) room. Divisional separation is achieved by locating the two shutdown divisions, which are made up of the respective shutdown equipment including, but not limited to, cabling and components, north and south of Column Line 17 which bisects the plant including the containment. Each fire area is enclosed in 3-hour-rated fire barriers. The 3-hour-rated fire barrier walls are located along the Column Line 17, except at elevations 115+6 feet and 130+6 feet, where the control room is located. Therefore, for a fire outside the containment, in any fire area other than the control room and the RSP room, at least one shutdown division remains free of fire damage, and safe shutdown can be achieved and maintained from the control room using the unaffected shutdown division.

As stated above, divisional separation does not exist between the two redundant shutdown divisions in the control room and the RSP room, both of which have redundant control function capability. The redundant shutdown capabilities in these rooms are physically separated and electrically isolated from each other. A fire in the RSP room will affect only the redundant control function capability in the RSP room. Therefore, in the event of a fire in the RSP room, safe shutdown can be achieved and maintained from the control room using either of the two shutdown divisions.

9.5.1.2.1.2 Fire in the Control Room

The System 80+ design includes alternative shutdown capability to achieve and maintain safe-shutdown following a disabling fire in the control room which requires control room evacuation. Current NRC regulations require that the transfer of control of the required safe-shutdown equipment to the RSP be independent of the fire area (control room). Current plants achieve this from outside the control room at such areas as the local equipment panels. The System 80+ design has the capability to transfer control of needed shutdown equipment from the control room to the RSP by utilizing two redundant sets of six transfer switches inside the control room. In addition, ABB-CE has installed transfer switches in channelized instrument and control equipment rooms. The staff finds this arrangement to be acceptable, as discussed below.

By letter dated January 7, 1994, and CESSAR-DC Section 7.4.1.1.10, ABB-CE indicated that the operator will transfer control of the required shutdown equipment to the RSP before leaving the control room. For this purpose, two redundant sets of six transfer switches are inside the control room, one set near each of the two control room exits located approximately 15.2 m (50 ft) apart. There are two shutdown divisions, as stated above, and each shutdown division has two channels for the shutdown equipment in the division. Each of these four channels has a separate transfer switch. The two remaining transfer switches are for each of the two process control system channels which are not related to safe-shutdown capability. The System 80+ design uses fiber-optic transfer switches and fiber-optic transmission cables. Actuation of the transfer switches at either control room exit initiates a "soft" transfer to deactivate the control room panel and activate the RSP for each of the six channels. Transfer switches are arranged so that when control capability is transferred from the control room to the RSP, manual operations in all six channelized instrument and control equipment rooms are required to return the control capability to the control room. Therefore, associated circuit interaction in the control room subsequent to transfer of control of shutdown capability from the control room to the RSP will not affect the ability to achieve and maintain safe shutdown from the RSP.

In addition to the control transfer capability that exists inside the control room, the System 80+ design provides the capability to transfer control of needed shutdown equipment from the control room to the RSP by utilizing transfer switches located in the channelized instrument and control equipment rooms. By letter dated January 7, 1994, ABB-CE indicated that in the unlikely event, the control room is evacuated before the transfer task is completed in the room, control can be transferred from the channelized equipment rooms. There are six of these rooms separated from one another by 3-hour-rated fire barriers, each room containing one of the six channel switches mentioned above. To account for the contingency of LOOP, the System 80+ design includes one and half-hour emergency lighting (battery powered) for the equipment room transfer switches and the pathways to these switches. (For information on 8-hour emergency lighting provided elsewhere to facilitate performance of safe shutdown following a fire, see Section 9.5.1.4.1 of this report.) The staff notes that the one and half-hour emergency lighting provision deviates from BTP CMEB 9.5-1 which states that eight hours of emergency lighting (battery powered) should be provided in all areas needed for operation of safe-shutdown equipment and in access and egress routes to and from these areas. However, in its discussion with the staff, ABB-CE indicated that the transfer task could be performed within 15 minutes. The staff conservatively

estimates that 45 minutes may be required to complete the transfer task after a disabling control room fire and loss of offsite power occur. Therefore, the staff finds the provision of one and half-hour emergency lighting for the transfer switches located in the channelized equipment rooms and the paths to these switches is acceptable.

In GL 86-10, Section 3.8.4, the staff states that the only manual action in the control room that can be credited in any safe-shutdown analysis for control room fire is scrambling the reactor from the control room before evacuating the control room. The subject section further states that any other manual action inside the control room deemed necessary before evacuation can be credited in the safe-shutdown analysis, provided it can be demonstrated that such action is capable of being performed before control room evacuation and that the effect of such an action would not be negated by subsequent spurious actuation signals arising in the control room from the postulated fire. In its shutdown analysis, besides taking credit for scrambling the reactor from the control room before evacuating the control room following a disabling control room fire, ABB-CE takes credit for transferring shutdown equipment control capability from the control room to the RSP by operating one set of transfer switches in the control room, before leaving the control room. By letter dated January 7, 1994, ABB-CE justified the credit taken in its shutdown analysis for performing the transfer task in the control room, before evacuating the control room, stating the following:

- (1) Avoidance of material that can ignite, explode, or support combustion in the design of the main control panels (MCPs); limits on energy sources coming into the MCPs to eliminate effectively potential ignition sources; current limits on power supplies and the use of light emitting diodes (LEDs) and low voltage interfaces between control switches and panel multiplexers to eliminate effectively fault-generated ignition sources; interfacing the control and indication signals with the MCPs via fiber optic cables in lieu of bringing high-energy control circuits into the MCPs, which is the control room panel conventional design; and reducing the number of indicators, switches, and other devices mounted on the control panel to reduce combustible loading, cumulatively result in absence of an energy source with sufficient energy to start a fire within the transfer switches themselves. Consequently, any fire that can affect the switches has to be an exposure (external) fire.
- (2) Since the redundant sets of transfer switches are widely separated, it is highly unlikely that an exposure fire would exist at both exits simulta-

neously. Therefore, the transfer task could be performed at the unaffected location of the transfer switches.

- (3) Since fiber optic switches and fiber optic cables are used in the System 80+ design, the worst that could happen from a fire that damages the switches would be the transfer of control of the affected channels from the control room to the RSP as a result of loss of the light signals (i.e., a fail-safe design).

However, the staff's position is that should a disabling fire occur in the control room, it is not known when the operators will evacuate should the need arise. It is not known when the smoke may obscure the operators' visibility and at what time toxic gases or oxygen-depleted air will force the operators from the control room. Because of these conditions, it is not certain that the operators will be able to transfer control to the remote shutdown panel before evacuating the control room. The staff expects that the operators will perform actions in the control room to support safe shutdown of the reactor including transfer of control to the remote shutdown panel before evacuating the control room if possible. However, due to the uncertainties previously identified, it is the staff's position that the independent method of transferring control to the RSP from the channelized instrument and control equipment rooms described above should be provided and maintained throughout the life of the plant.

For the reasons stated above, the staff concludes that the System 80+ design has the capability to transfer control from the control room to the RSP. The staff further notes that in the event control is transferred before evacuating the control room, any associated circuit interaction in the control room subsequent to the completion of the transfer task in the control room, will not compromise the ability to achieve and maintain safe shutdown from the RSP. Additionally, as discussed above, the staff concludes that the System 80+ design provides control transfer capability in the channelized instrument and control equipment rooms which can be used in the event the control room is evacuated before control is transferred in that room. Furthermore, in CESSAR-DC Section 9.5.1.2, ABB-CE identifies a COL action item (item A) which calls for the COL applicant to provide procedures and training for transferring control from the control room to the RSP from either the transfer switches located in the control room or located in the maintenance and test panels in the channelized instrument and control equipment rooms. On this basis, the staff concludes that the System 80+ design complies with the guidance of BTP CMEB 9.5-1 and is, therefore, acceptable.

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9.5.1.2.1.3 Protection of Safe-Shutdown Capability for a Fire Inside the Containment

In the System 80+ design, the containment and annulus constitute a single fire area. The only equipment inside the containment and annulus which are required for safe-shutdown are motor-operated valves (MOVs), instruments associated with safe-shutdown divisions, and shutdown cabling. The staff recognizes the need for open communication between compartments inside the containment in order to be able to relieve and equalize pressure following a high-energy line break. Therefore, the staff has stated in SECY-90-016 that evolutionary ALWRs should provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. To meet this criterion, in CESSAR-DC Section 9.5.1.2, ABB-CE states that the fire protection safe-shutdown analysis (which will be maintained as part of the System 80+ design basis) will ensure that fire at any specific location inside the containment will not affect the redundant safe-shutdown components. ABB-CE further states that redundant safe-shutdown components such as instruments and valves, will be separated to the extent practicable as stipulated in SECY-90-016. In CESSAR-DC Section 9.5.1.3.8, ABB-CE elaborates the separation criterion by stating that redundant trains of valves and instruments analyzed as an assured method of achieving safe shutdown will be physically separated by reinforced-concrete walls, pressure vessels such as the pressurizer or steam generator, or by spatial separation (i.e., greater than 6.1 m [20 ft]) with no intervening combustibles, so that a potential fire will not affect redundant equipment. In CESSAR-DC Section 9.5.1.3.9, ABB-CE discusses in detail how redundant safe-shutdown functions and associated equipment are protected against a fire either inside or outside the containment. On the basis of its review of this material, the staff agrees with ABB-CE that redundant safe-shutdown valves and instruments will not be affected by a fire inside the containment or annulus.

Regarding cables used for safe-shutdown functions inside the containment and annulus, ABB-CE states in CESSAR-DC Sections 9.5.1.2, Item B and 9.5.1.3.8 that these will be three-hour rated fire cable protective systems (i.e., mineral-insulated cables) with one exception. The exception to the three-hour fire-resistance rating may be the containment penetrations (installed in the containment vessel to facilitate transition between inner and outer containment) for these cables, which are currently only available with a 1-hour fire-resistance rating. However, the standard design includes a commitment to purchase 3-hour-rated fire penetrations, if they become available.

In the earlier (preceding the issuance of the DSER) version of CESSAR-DC Section 9.5.1, ABB-CE proposed using mineral-insulated cables that would qualify as either a 3-hour-rated fire barrier or a radiant energy heat shield. When these mineral-insulated cables would be used as a radiant energy shield, the installation would ensure a minimum separation between redundant shutdown divisions of 6.1 m (20 ft) with no intervening combustibles. Though the staff recognized the need for certain redundant components to converge inside the containment, the staff found the concept of radiant energy heat shields and 6.1 m (20-ft) separation, unacceptable. Therefore, the staff asked ABB-CE to justify fully each such deviation inside the containment, and designated this as DSER Open Item 9.5.1.2.1-1. In response to the concern, ABB-CE has revised CESSAR-DC Sections 9.5.1.2, Item B and 9.5.1.3.8. As discussed above, ABB-CE deleted the use of mineral-insulated cables as radiant energy heat shields. ABB-CE further states that the cables will qualify as 3-hour-rated fire barriers. Additionally, as discussed in the first paragraph of this Section (9.5.1.2.1.3), in CESSAR-DC Section 9.5.1.3.8, ABB-CE elaborates on how redundant safe-shutdown equipment, such as instruments and valves, will be separated to the extent practicable as stipulated in SECY-90-016. To provide added assurance regarding separation of redundant safe-shutdown equipment, ABB-CE identifies a COL action item in CESSAR-DC Section 9.5.1.12 which calls for the COL applicant to complete a plant-specific fire hazards analysis to include, but not be limited to, a plant-specific safe-shutdown analysis. The plant-specific fire hazard analysis will use the Fire Hazards Assessment prepared by ABB-CE for the System 80+ design, as the base document. The staff will, therefore, review the plant-specific safe-shutdown analysis which, in turn, will address among other things, the specific separation provided for redundant individual safe-shutdown equipment inside the containment and annulus, for each COL applicant, separately. For these reasons, the staff concludes that the System 80+ design complies with the intent of SECY-90-016 regarding the fire-protection separation criterion for redundant shutdown equipment inside the containment and annulus. On this basis, DSER Open Item 9.5.1.2.1-1, is resolved.

9.5.1.2.1.4 Additional Design Features To Protect Safe-Shutdown Capability

In the DSER, the staff stated that Section 4.1.2, "Inside Containment and Annulus" of the Fire Hazards Assessment suggested that four shutdown trains should be provided for the containment and annulus area, with the redundant trains being separated by quadrants to provide sufficient spatial separation between the trains. The staff further stated that Section 7.6, "Associated Circuits," of the Fire Hazards Assessment reinforced the general design feature

of four shutdown trains in the System 80+ design by stating that the transfer switches for each of the four channels of safety-related electrical equipment are separated by 3-hour rated fire barriers. The staff, however, found the division and channel listing of safe-shutdown equipment in Section 7.7 "Redundant Fire Areas Containing Safe Shutdown Equipment," of the Fire Hazards Assessment confusing. For these reasons, the staff stated that ABB-CE should clarify the independence, separation, fire barriers, and other fire-protection features for redundant shutdown trains in the System 80+ design, and designated this as DSER Open Item 9.5.1.2.1-2.

In response to this staff concern, ABB-CE revised Section 4.1.2 of the Fire Hazards Assessment. As revised, Section 4.1.2 no longer states that redundant shutdown divisions inside the containment and annulus are separated by quadrants. Instead, it summarizes the information on protection of safe-shutdown capability for a fire inside the containment and annulus provided above (Section 9.5.1.2.1.3 of this report). As discussed above, the staff finds the protection in the containment and annulus acceptable. Regarding the 3-hour-rated fire barrier separation between channelized transfer switches referred to in Section 7.6 of the Fire Hazards Assessment, in a meeting with the staff on December 14, 1993, ABB-CE clarified that the transfer switches referred to in the subject section are the channel switches for transferring control of safe-shutdown equipment from the control room to the RSP. One switch is located in each of the four channelized instrument and control equipment rooms. The staff notes that by virtue of the fact that these rooms outside the containment are separated from one another by 3-hour-rated fire barriers, the associated channelized transfer switches located in these rooms are also separated from one another by 3-hour-rated fire barriers. Additionally, ABB-CE has revised Section 7.6 of the Fire Hazards Assessment, deleting the subject reference.

Regarding the division and channel listing of the shutdown equipment in Section 7.7 of the Fire Hazards Assessment, which the staff stated was confusing (in the DSER), ABB-CE has submitted information about these additional design features (CESSAR-DC Section 9.5.1.14) to explain the listing in the subject section. These additional design features provide separation of shutdown equipment, not only between divisions, but also between redundant shutdown equipment within a division (i.e., channel separation) outside the containment. The staff recognizes that so far as safe-shutdown capability for an evolutionary ALWR is concerned, it is sufficient to show that following a fire in any fire area, at least one division of safe-shutdown equipment free of fire damage is available for achieving and maintaining safe shutdown without the need for re-entry into the fire area for repairs and operator

actions to achieve and maintain safe shutdown. As discussed above, the System 80+ design meets this criterion for a fire in any fire area outside the containment, control room, and RSP room by three-hour-rated fire barrier separation between redundant shutdown divisions. In the event of a fire in the control room, the RSP, which is electrically and physically independent of the control room, provides alternative shutdown capability. Conversely, in the event of a fire in the RSP room, the control room remains available and is not affected. Therefore, normal shutdown capability is provided in the control room. Further, as discussed above, though the 3-hour-rated fire barrier separation does not exist between redundant divisional shutdown equipment inside the containment and annulus, there is sufficient separation between redundant equipment in the containment and annulus, to assure the above criterion. Therefore, the System 80+ design conforms to the BTP CMEB 9.5-1 criterion as well as to the SECY-90-016 position with respect to ensuring that at least one division of safe-shutdown equipment, free of fire damage, is available for achieving and maintaining safe shutdown following a fire in any plant fire area.

In CESSAR-DC Section 9.5.1.14, ABB-CE addresses a staff concern by adding a detailed discussion on design features beyond the divisional separation, provided outside the containment to reduce the risk of a core melt from a fire outside the containment. As an example, in the subject section, ABB-CE points out that the reactor building subsphere (which contains some fire areas outside the containment) is divided into quadrants and that both the divisional wall and quadrant wall within a division are 3-hour-rated fire barriers and that redundant shutdown equipment within a division (e.g., one shutdown cooling pump and one safety injection pump; and the corresponding redundant equipment which are one containment spray pump and another safety injection pump) are separated by quadrant walls. The switchgear associated with the quadrants are also separated by 3-hour-rated fire barriers and the cables from the switchgear are routed through separate fire areas within the reactor building subsphere. The section also explains how channel separation within a shutdown division is provided outside the reactor building subsphere for channelized equipment within a division. Having reviewed the section which includes an exception to the design feature of channel separation within a division (in the DG room, cables from each DG to its respective Class 1E switchgear are located in the same fire area) and compensatory design features provided for the exception, the staff finds that ABB-CE has satisfactorily addressed the staff's concern relating to divisional and channel separation of redundant equipment outside the containment.

On this basis DSER Open Item 9.5.1.2.1-2 is resolved.

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In CESSAR-DC Section 9.5.1.3.9, ABB-CE discusses in detail how the System 80+ design protects the redundant safe-shutdown functions provided by the redundant shutdown equipment against the effects of a fire either inside or outside the containment. The safe-shutdown functions are maintaining RCS pressure boundary integrity; ensuring proper reactivity control; making available reactor coolant makeup to maintain reactor coolant in the pressurizer within prescribed levels; maintaining needed RCS decay heat removal, ensuring depressurization to cooldown the RCS to safe-shutdown conditions; providing direct readings of needed process variables; maintaining support functions; and ensuring availability of onsite electrical power which is needed to power safe-shutdown equipment should offsite power be lost. The discussion addresses how the performance of these safe-shutdown functions are not compromised by spurious operations induced by a fire either inside or outside the containment. Specifically, the section indicates that adverse effects due to fire-induced spurious operations are prevented by one of, or an applicable combination of, the following design features: (1) needed shutdown system lines have two power-operated valves in series with the valves powered by different divisions or different channels within a division, (2) the valves are widely separated and in different fire areas, (3) the associated MCCs for the valves are in different fire areas, and (4) the MCC breakers associated with the valves are opened once the valves are placed in desired position (i.e., closed or open). In the section, ABB-CE also states that the solenoid valve power supply fuses are normally removed to prevent fire-induced spurious opening of the single isolation valves on each of the two vent lines of each safety injection tank. ABB-CE has also submitted a Fire Hazards Assessment document to the NRC. This document, among other things, includes a safe-shutdown analysis for the System 80+ design. In the document, ABB-CE listed or discusses, as appropriate, the following: (1) the criteria for achieving and maintaining safe shutdown following a fire (i.e., the ability to achieve and maintain safe shutdown without entering into the fire area for repairs or manual operations), (2) design-basis goals for safe shutdown, (3) safe-shutdown performance objectives, (4) systems required for safe shutdown, (5) safe-shutdown components, (6) protection against associated circuit concerns, (7) prevention of fire-induced high/low-pressure interface breaches, and (8) a list of fire areas that contain equipment required for safe shutdown following a fire and the redundant areas that contain the corresponding redundant equipment. Regarding preventing fire-induced high/low-pressure interface breaches, the subject document (Section 7.6) states that the RCS MOVs which serve as high/low-pressure interfaces and are required to be closed during normal power operation, have the valve motors deenergized during power operation to prevent such fire-induced breaches.

On the basis of its review of the Fire Hazards Assessment document and CESSAR-DC Sections 9.5.1.3.6, 9.5.1.3.7, and 9.5.1.3.8, the staff concludes that associated circuit interactions due to a fire in any plant fire area will not affect the capability to achieve and maintain safe shutdown because of the following features: (1) complete separation of redundant shutdown divisional equipment which include their respective power and control circuits by 3-hour-rated fire barriers for all fire areas outside the containment, control room, or RSP room; (2) physical and electrical independence between alternative shutdown capability provided at the RSP and the normal shutdown capability provided in the control room; and (3) use of 3-hour-rated fire cable protective systems (mineral insulated cables) inside the containment and annulus for cables associated with safe-shutdown functions. To provide added assurance regarding separation of redundant safe-shutdown divisions and consequently safe-shutdown capability, ABB-CE has designated a COL action item in CESSAR-DC Section 9.5.1.12 which calls for the COL applicant to complete a plant-specific fire-hazards analysis. In CESSAR-DC Section 9.5.1.3.6, ABB-CE states that this analysis will include a finalized associated circuits study. Therefore, each COL applicant will submit a plant-specific associated circuits study in its plant-specific fire hazards analysis.

In CESSAR-DC Section 9.5.1.3.4, ABB-CE states that shutdown procedures following a fire are the same as described in the plant emergency procedures for achieving safe shutdown (i.e., cold shutdown) and do not require additional personnel for the fire scenario. Specifically, ABB-CE states that cold shutdown can be achieved with one shutdown division within 36 hours after reactor trip. The staff finds the time limit of 36 hours acceptable based on the criteria of BTP CMEB 9.5-1 and SECY-90-016 which indicate that cold shutdown is to be achieved within 72 hours and that cold shutdown conditions are to be maintained thereafter when utilizing alternative dedicated shutdown capability.

9.5.1.2.1.5 Conclusion on Safe-Shutdown Capability

On the basis of this discussion, the staff concludes that System 80+ design has the capability to achieve cold shutdown within 72 hours after the reactor trip following a fire event anywhere in the plant and maintain the safe shutdown thereafter, without relying on re-entry into the fire area for any repair or manual operation. The staff further concludes that the System 80+ design complies with the guidelines of BTP CMEB 9.5-1 and SECY-90-016 with regard to safe-shutdown capability following a fire event in the plant and is, therefore, acceptable. However, as stated above, the staff will review the COL applicant's plant-specific safe-shutdown analysis which, in turn, will

include, but is not limited to, review of specific separation provided for redundant individual safe-shutdown equipment in the containment and annulus, and the finalized associated circuits study involving fire areas both inside and outside the containment and annulus, as part of its review of plant-specific fire hazards analysis.

9.5.1.2.2 Passive Fire-Protection Features

In the DSER, the staff indicated that ABB-CE had made only broad statements concerning building assemblies and other construction-related, passive, fire-protection features for the ABB-CE System 80+ design, such as walls, partitions, floor-to-ceiling assemblies, columns, beams, and doors; insulating materials, such as cable wraps and heat-resistant coatings; and such penetrations through the building assemblies as doorways, hoistways, stairways, and cable trays and conduits. This was designated as DSER Open Item 9.5.1.2.2-1.

In a letter dated June 11, 1993, (LD-93-090), ABB-CE stated that CESSAR-DC Section 9.5.1.2 would be modified to state that 3-hour-rated fire barriers will be installed to separate redundant divisions outside the containment with the exception of the control room and the remote shutdown panel room. Also ABB-CE stated that CESSAR-DC Section 9.5.1.3.2.2 would be modified to state that walls, floors, and ceiling assemblies designated as fire barriers conform to the acceptance criteria of American Society for Testing and Materials (ASTM) E-119, "Fire Tests of Building and Construction Materials."

ABB-CE's proposal to provide 3-hour-rated fire barriers to separate redundant divisions complies with the guidelines of BTP CMEB 9.5-1 and SECY-90-016 and is acceptable. On this basis, DSER Open Item 9.5.1.2.2-1 is resolved. The COL applicant's maintenance program will ensure such that fire-rated assemblies as fire doors, fire dampers, and penetration seals are maintained as required by their respective NFPA codes or manufacturer's instructions.

ABB-CE has paid particular attention to ventilation paths in the System 80+ design. Each division has its own HVAC system. This means that ventilation air supply, return, and exhaust for any division are independent of all other divisions, and, with two exceptions, HVAC ducting does not penetrate three-hour-rated fire barriers between divisions.

One exception to the division-specific HVAC system is a single opening in the divisional fire wall that separates the redundant AHUs. An air-intake duct that supplies makeup air to the redundant control room system passes through this opening. The opening is protected with a combination fire and smoke damper. This arrangement is necessary for

nuclear safety reasons to ensure acceptable control room doses (refer to Section 6.4 of this report). The control room habitability system will be designed to meet the single failure criteria having dual air intakes, filtration and cooling systems. The other exception is the FBVS.

In the DSER, the staff identified the need for a description of the design and operation of the components used in the smoke removal mode of operation. This was designated as DSER Open Item 9.5.2.2-2.

In CESSAR-DC, Section 9.5.1.2, ABB-CE states that the HVAC system is designed to remove smoke and mitigate smoke migration beyond the area of origin in the event of fire. The dedicated fans for smoke purge are designed to exhaust at a minimum of 945 L/minute/m² (3 cfm/ft²) of floor area. The normal ventilation is designed to provide an air flow of 315 L/minute/m² (1 cfm/ft²) of floor area or more. ABB-CE states that the layout of the ductwork ensures ventilation of all corners of the area as much as practical. The design, as described, provides a lower pressure into the division experiencing the fire that will prevent or significantly reduce the amount of smoke migration to other divisions. In Amendment U to CESSAR-DC, Section 9.5.1.8.2, ABB-CE stated that the ventilation system is designed in accordance with NFPA Standard 92B, "Guide for Smoke Management Systems in Malls, Atriums and Large Areas." ABB-CE's proposed HVAC design complies with the guidance of BTP CMEB 9.5-1 and SECY-90-016, and is acceptable. On this basis, DSER Open Item 9.5.1.2.2-2 is resolved.

9.5.1.3 Fire-Protection System

9.5.1.3.1 Fire Detection

The automatic fire detection systems will be designed and installed in accordance with NFPA Standard 72; they will be provided for all significant hazards and safe-shutdown components.

In the DSER, the staff stated that although most details in the Fire Hazards Assessment concerning installation of the various fire-detection systems are indicated as TBD (to be determined), the staff makes certain assumptions regarding the design and installation of fire-detection capability in the ABB-CE System 80+ design. The staff assumes that ABB-CE provides detection capability for major cable concentrations, safe-shutdown-related major pumps, switchgear, motor-control centers, battery and inverter areas, relay rooms, fuel areas, and all other areas containing appreciable in situ or potentially transient combustibles. Detector devices will be selected on the basis of type of anticipated fire and will be located on the basis of ventilation, ceiling height, ambient conditions, and burning

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characteristics of the involved materials. Detection systems will alarm and annunciate in the control room and will give a distinctive audible and, if necessary (to facilitate fire brigade identification of fire location), visual local alarm. Verification of this assumption was designated as DSER Open Item 9.5.1.3.1-1.

In Amendment O to CESSAR-DC Section 9.5.1.7.6, ABB-CE indicated that fire detection will be installed in areas containing major cable concentrations, safe-shutdown major pumps, switchgear, motor control centers, battery and inverter areas, relay rooms, fuel areas and areas containing appreciable in situ or potentially transient combustible materials. The staff finds that ABB-CE's detector design complies with the guidance of BTP CMEB 9.5-1 and SECY-90-016 and is, therefore, acceptable.

9.5.1.3.2 Fire-Protection Water-Supply System

In the DSER, the staff stated that ABB-CE did not discuss the fire-protection water-supply system in the Fire Hazards Assessment. In Section 5.1 of that assessment, "Defense-in-Depth," ABB-CE stated that failure of one water source or of any portion of the exterior fire-water distribution system does not affect delivery of an adequate water supply to any fixed fire-suppression system. This was the only reference in the Fire Hazards Assessment to a fire-protection water-supply system. Also, the discussion in CESSAR-DC Section 9.5.1.5.2 (Amendment I), was not sufficiently detailed for the staff to complete its review. This was designated as DSER Open Item 9.5.1.3.2-1.

In Amendment U to CESSAR-DC Section 9.5.1.7, ABB-CE stated that each fire pump is of the correct size to produce both the maximum water volume and system pressure demand with the shortest flow path out of service. ABB-CE also stated that sectional isolation valves are located throughout the water distribution system to ensure that any portion of the distribution system that serves buildings containing safety-related systems, equipment, and components can be repaired without isolating primary and secondary fire protection. ABB-CE's design for the dedicated fire protection water supply and distribution system complies with the guidance of BTP CMEB 9.5-1, Section C.6.b, relating to fire protection water supply systems, and is therefore acceptable. On this basis DSER Open Item 9.5.1.3.2-1 is resolved.

In the DSER, the staff stated that the following are the minimum acceptable criteria for a fire-protection water-supply system:

- (1) A dedicated fire-protection water-supply and distribution system must be designed and

installed in accordance with NFPA Standards 11, 13, 14, 15, 20, and 24 to meet the anticipated needs for fixed water-suppression systems and manual hose stations.

In CESSAR-DC Section 9.5.1.7, ABB-CE stated that the fire-protection water-supply and distribution system is designed and will be installed according to NFPA standards. ABB-CE's commitment to design the system in accordance with NFPA standards complies with the guidance of BTP CMEB 9.5-1 and is, therefore, acceptable.

- (2) The sprinkler systems in the reactor building and the wet standpipe systems in the reactor and control buildings must be designed in accordance with ANSI B31.1 "Power Piping" and analyzed to remain functional following a SSE. A portion of the water-supply system (a tank, a pump, and part of the yard supply main) must also be designed to these requirements. The remainder of the water systems must be designed to the appropriate fire protection standards such as NFPA 13 for "Installation of Automatic Sprinkler Systems" and NFPA 15 for "Water Spray Fixed Systems". During normal operation, the seismically designed and non-seismically designed systems must be separated by normally closed valves and a check valve, so that a break in the non-seismically analyzed portion of the system cannot impair the operation of the seismically designed portion of the system.

In CESSAR-DC Section 9.5.1.7.3.C, ABB-CE stated that the sprinkler system piping is seismically restrained to avoid interaction with systems, equipment, and components that must function following a design-basis seismic event. Also, in Section 9.5.1.7.4, ABB-CE states that fire hose and standpipe systems located in the reactor building and nuclear annex are designed to remain functional following the design-basis earthquake. Each connection of the standpipe system to the fire-protection water-distribution system includes a manual isolation and a back-flow-prevention check valve which is seismically qualified.

The fire hose and standpipe systems located in the reactor building and the nuclear annex will be designed to remain functional following a SSE. The piping system serving such hose stations will be analyzed for SSE loading and has supports to ensure the integrity of system pressure. The piping

and valves for the portion of the hose standpipe system affected by this functional requirement will be designed, as a minimum, to comply with ANSI B31.1 as indicated in BTP CMEB 9.5.1. The System 80+ design, as discussed, complies with the guidance of BTP CMEB 9.5-1 Section C.6.c, and, therefore, is acceptable.

- (3) The water-supply system must furnish fresh water, filtered if necessary, to remove silt and other debris. Two sources, with a minimum capacity of 1.1×10^6 L (300,000 gal) for each source, must be provided. If the primary source is a volume-limited supply, such as a tank, a minimum of 450,000 L (120,000 gal) must be passively reserved for use by the seismically designed portion of the suppression system. This reserve will supply two manual hose reels for two hours. The train must be designed to remain functional following the SSE. Motive power for the pump in this train can be furnished by a diesel engine or by electric power supplied from a non-Class 1E bus which is fed by one of the EDGs. A jockey pump must be provided to keep the system pressurized.

In CESSAR-DC Section 9.5.1.7.1, ABB-CE states that the fire-protection water supply has two 1.1×10^6 L (300,000 gal) water systems with an automatic fill system supplied from the treated water system. In Section 9.5.1.7.4, ABB-CE states that a seismically designed pump and seismically designed storage tank with a capacity of 68,000 L (18,000 gal) which will supply two fire hoses delivering water at 300 L/minute (75 gpm) at a minimum of 450 kPa (65 psi) for 2 hours.

ABB-CE's design as discussed complies with the guidance of BTP CMEB 9.5-1, Section C.6.b relating to fire-protection water supply systems and is, therefore, acceptable.

In the DSER, the staff stated that the turbine building must have modified Class III standpipes, hose reels, and ABC portable extinguishers throughout the building. In addition, the following fire-suppression systems must provide primary fire-suppression capability to the following areas:

- automatic closed-head sprinkler systems in the open grating area of the three floors under the turbine

- automatic sprinklers or deluge foam-water sprinkler systems in the lube oil handling area and the lube oil reservoir area
- a deluge sprinkler system in the hydrogen seal oil unit area
- a sprinkler system in the auxiliary boiler area

Fire-suppression systems for the turbine building may receive water from the portion of the supply system that is not required to be seismically analyzed for SSE.

The main power, unit auxiliary, and reserve transformers must have deluge-water-spray suppression systems. The systems must be automatically actuated by flame or temperature detectors. An oil and water collection pit must be beneath each transformer. Each pit must have drains sloping away from buildings and transformers. Shadow-type fire-barrier walls must be between adjacent transformers.

Alarm systems, both manual and automatic, must be in all areas of the plant as passive systems. They alarm without controlling an extinguishing function.

At least two fire pumps must be located in separate fire areas cut off from each other and from the rest of the plant by 3-hour-rated fire barriers.

The fire-main loop in the yard must be designed and installed with sectional control valves that will deliver total fire flow to all automatic and manual fire-suppression systems and manual hose stations, even if the shortest portion of the water distribution piping is out of service.

Pending confirmation of ABB-CE's compliance with the guidelines of Section C.6.b of BTP CMEB 9.5-1 pertaining to fire systems, this was designated as DSER Open Item 9.5.1.3.2-1.

In the letter dated June 11, 1993 (LD-93-090), ABB-CE stated that the fire-protection features for the turbine building, the main power, unit auxiliary, and the reserve transformers will be provided as necessary to meet the overall fire-protection goals and objectives which include, but are not limited to, the following:

- (1) Prevent loss of ability to achieve safe shutdown following a fire.
- (2) Prevent fire from threatening more than any one division of equipment or components required to achieve cold shutdown.

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- (3) Prevent fire from damaging more than one division of safety-related structures, equipment, or components.

ABB-CE stated that fire-protection features will be selected consistent with guidance in NFPA Standard 803 and the requirements of property insurers. The type of fire-protection to be provided for the turbine building and the other areas specified above will be determined by the fire hazards analysis performed by the COL applicant. As discussed in Section 9.5.1.6 of this report, the staff will review the acceptability of the fire hazards analysis.

9.5.1.3.3 Water Fire-Suppression Systems

Automatic water and foam fire-suppression systems will be installed above major fire hazards to be identified by the fire hazards analysis. The systems are designed and will be installed in accordance with NFPA Standards 11, 13, and 15.

Standpipe and hose stations will be installed throughout the plant on the basis of needs identified in the fire hazards analysis. The standpipe systems are designed and will be installed in accordance with NFPA Standard 14. Each station is equipped with a suitable hose and an on/off spray nozzle that are listed or approved by a nationally recognized testing laboratory.

In the DSER, the staff stated that details concerning possible need for pressure-reducing orifices, exterior hydrants and hose houses, and electrical supervision of control and sectionalizing valves were not discussed in the Fire Hazards Assessment. These were designated as DSER Open Item 9.5.1.3.3-1.

ABB-CE revised CESSAR-DC Section 9.5.1.7.4 to state that "Connections have pressure reducing orifices if necessary to maintain a maximum system pressure at 690 kPa (100 psi) for fire fighter safety." Also in CESSAR-DC Section 9.5.1.7.2, ABB-CE indicates that fire hydrants are located approximately 76 m (250 ft) apart and the hydrants have individual isolation valves so that they can be individually isolated for repair. Sectional isolation valves are located throughout the water distribution system to ensure that any portion of the distribution system that serves buildings containing safety-related systems, equipment, and components can be repaired without isolating primary and secondary fire protection. ABB-CE indicated that the fire-protection control valves are either locked or electrically supervised to ensure that they remain in the open position. Administrative controls prevent the use of the fire-protection water for other purposes. The water distribution system, hydrants, and hose houses will be installed according to NFPA Standard

24. On this basis, DSER Open Item 9.5.1.3.3-1 is resolved.

In the DSER, the staff stated that ABB-CE must also confirm that there are no penetrations through fire barriers separating fire areas containing redundant trains of safe-shutdown equipment and that no possibility exists for channeling water from fire-extinguishing operations in one redundant fire area into another redundant fire area. This was designated as DSER Open Item 9.5.1.3.3-2.

In the letter dated June 11, 1993 (LD-93-090), ABB-CE stated that internal flood protection protects safe-shutdown equipment as described in CESSAR-DC Section 3.4.4.1. The staff reviewed the flood protection features in Section 3.4.1 of this report and finds that they are designed to protect safe-shutdown equipment from flooding, including flooding from fire-suppression activities. On this basis, DSER Open Item 9.5.1.3-1 is resolved.

Flood protection of safe-shutdown equipment complies with the guidelines of BTP CMEB 9.5-1 Section C.1.c relating to fire suppression systems design basis and, therefore, is acceptable.

9.5.1.3.4 Gaseous Fire-Suppression Systems

ABB-CE did not discuss gaseous fire-suppression systems in the Fire Hazards Assessment for the System 80+ design. Therefore, the staff assumes that no gaseous fire-suppression systems will be installed in the ABB-CE plant.

9.5.1.3.5 Fire Extinguishers

Portable fire extinguishers are provided in areas with in situ combustibles or potentially transient combustibles. Extinguishers are chosen on the basis of the anticipated type of fire in the area and the effect of the extinguishing agent on equipment in the area. The portable extinguishers are selected, installed, and maintained according to the provisions of NFPA Standard 10. This conforms to the guidelines of Section C.6.f of BTP CMEB 9.5-1 and, therefore, is acceptable.

9.5.1.4 Fire-Protection Support Systems

9.5.1.4.1 Emergency Communication and Lighting

In Section 1.4.3 of the Fire Hazards Assessment, ABB-CE stated that "Communication equipment and lighting is available to facilitate fire control and suppression and mitigate damage." In Section 5.3.8, ABB-CE stated that "Emergency lighting is provided for personnel egress and fire brigade access to each fire area." In the DSER, the staff stated that additional information would be required

to complete its review. This was designated as DSER Open Item 9.5.1.4.1-1.

In CESSAR-DC Section 9.5.1.6.6 (Amendment J), ABB-CE stated that a radio system with a frequency dedicated for fire brigade use will be provided. Dedicated radios with their respective charger base will be stored in the fire brigade storage room. In CESSAR-DC Section 9.5.1.8.1, ABB-CE indicates that 8-hour battery-powered emergency lighting will be provided for the remote shutdown panel, control room, and the pathway from the control room. Lighting for transfer switches and pathway to transfer switches located outside the control room are discussed in CESSAR-DC Section 9.5.1.2.1.2 and require a one and half-hour battery power supply. On the basis of this additional information, the staff finds that the emergency communications and emergency lights comply with guidance of BTP 9.5-1 Section C.5.g. relating to lighting and are, therefore, acceptable. On this basis, DSER Open Item 9.5.1.4.1-1 is resolved.

9.5.1.4.2 Emergency Breathing Air

In the DSER, the staff stated that ABB-CE did not discuss emergency breathing air in the Fire Hazards Assessment, as is recommended in SRP Section 9.5-1, C.3.c. This was identified as DSER Open Item 9.5.1.4.2-1. In the April 28 and 29, 1993, meeting with NRC, ABB-CE committed to comply with BTP CMEB 9.5-1. In a letter dated June 11, 1993 (LD-93-090), ABB-CE stated that CESSAR-DC 9.5.1.8.7 is consistent BTP CMEB 9.5-1. ABB-CE will provide emergency breathing air in accordance with BTP CMEB 9.5-1. On this basis, DSER Open Item 9.5.1.4.2-1 is resolved.

9.5.1.4.3 Curbs and Drains

In the DSER, the staff stated that the only reference to curbs and drains in the Fire Hazards Assessment is a single sentence under Section 4.2, "General Fire Protection Features." At the seventh bullet, ABB-CE stated: "Where fixed fire suppression systems are installed, provisions for control of water drainage and features to mitigate damage due to water discharge are included." This was not a sufficient discussion to permit staff evaluation. In addition, ABB-CE must address curbs and drains for areas in which only manual fire-fighting operations will occur, not just for areas with automatic fire-suppression systems. This was designated as DSER Open Item 9.5.1.4.3-1.

In a letter dated June 11, 1993 (LD-93-090), ABB-CE stated that "Curbs and drains are discussed in Section 9.5.1.6.4 of CESSAR-DC. Equipment required for safe shutdown that is susceptible to water damage will be protected with curbs and/or drains. Exact location and

sizing of drains and location of equipment curbs will be determined during detailed design."

In CESSAR-DC Section 9.5.1.8.4, ABB-CE states that where fixed fire-protection systems are installed, floor drains are provided, sized to collect water discharge. In areas where drains are not installed due to pressure boundary constraints, equipment susceptible to water damage is installed on 6-inch elevated curbs. On the basis of these statements, DSER Open Item 9.5.1.4.3-1 is resolved.

9.5.1.4.4 Smoke Control

In the DSER, the staff stated that ABB-CE must submit more detailed information on utilization of the HVAC system for smoke removal and smoke control during fire. This was designated as DSER Open Item 9.5.1.4.4-1. In CESSAR-DC Section 9.5.1.8.2, ABB-CE indicated that the ventilation systems are designed in accordance with NFPA Standard 90A "Air Conditioning and Ventilation Systems" and 92B, "Guide for Smoke Management Systems in Malls, Atriums, and Large Areas." As discussed in Section 9.5.1.2.2 of this report, ABB-CE's response is acceptable. On this basis, DSER Open Item 9.5.1.4.4-1 is resolved.

9.5.1.4.5 Access/Egress Routes

In Section 1.4.1 of the Fire Hazards Assessment, ABB-CE stated that the plant arrangement is carefully reviewed to ensure adequate means of personnel egress and fire brigade access are provided. Additionally, in Section 3.2 of the Fire Hazards Assessment, ABB-CE stated that it will comply with the guidance of SRP Section 9.5.1. The staff regards this as a commitment to provide clearly marked exit routes for each fire area. These routes are designed to comply with applicable life safety codes and standards. These provisions for access and egress routes conform to the guidelines in Section C.5.g of BTP CMEB 9.5-1 and Section III.G of Appendix R to 10 CFR Part 50, and are acceptable.

9.5.1.4.6 Construction Materials and Combustible Contents

ABB-CE commits in the System 80+ Design Fire Hazards Assessment (Section 1.4.1) to furnish appropriate fire-resistance ratings for structural members, and noncombustible or fire-retardant interior finish materials. ABB-CE also commits to comply with the provisions of SRP Section 9.5.1.

ABB-CE does not mention transformers located inside the plant. However, the staff understands that commitments

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also apply to transformers, and that dry-type transformers are located inside fire areas containing safety-related equipment.

These provisions, which comply with the intent of the guidelines in Sections C.5.a.9 and C.5.a.12 of BTP CMEB 9.5-1 to use only noncombustible materials for interior finish, are acceptable.

9.5.1.4.7 Interactions With Other Systems

In the DSER, the staff stated that ABB-CE was not clear regarding the protection of safe-shutdown trains and how they are separated. It was not certain, therefore, whether or not adequate protection has been provided for safe-shutdown equipment with respect to potential damage from water spray released during fire-suppression activities. This was designated as DSER Open Item 9.5.1.4.7-1.

In response, ABB-CE stated in CESSAR-DC, Amendment O, Section 9.5.1.8.3, that curbs, drains, conduit seals, equipment pedestals and water shields (inside containment) protect equipment required for safe shutdown from an inadvertent or advertent discharge of water from automatic or manual fire-fighting activities. ABB-CE stated that outside the containment, safe-shutdown equipment need not be shielded from manual fire-fighting operations since the safe-shutdown equipment is separated by 3-hour-rated fire barriers that will confine the fire and fire-fighting operations to a single area. ABB-CE also stated that safe-shutdown equipment need not be shielded from an inadvertent operation of the sprinkler system since the sprinkler system is a preaction system that requires the activation of the automatic fire detector and the activation of the sprinkler head before it releases any water. ABB-CE stated that shielding is provided within containment, as deemed necessary following interaction review during detailed design and as-built walkdowns. The open ends of all vertical conduit, and the open ends of all horizontal conduit that terminate within 18 inches of a floor, are sealed to keep water from infiltrating.

These provisions comply with the guidelines of Sections C.5.a.(14) and C.6.c.(1) of BTP CMEB 9.5-1 and are, therefore, acceptable. On this basis, DSER Open Item 9.5.1.4.7-1 is resolved.

9.5.1.4.8 Preoperational Testing

In CESSAR-DC Section 9.5.1.9, ABB-CE identifies the startup tests necessary to comply with the provisions of SRP Section 9.5.1 regarding preoperational testing. Therefore, all of the active components of the entire fire-protection system(s) pass a preoperational acceptance test

in accordance with the appropriate NFPA standard governing design and installation of the system. The following components and systems must pass the preoperational testing before being placed in service:

- fire pumps – controls, flow volume, and pressure
- water distribution – flush and hydrostatic water
- control valves
- fire-detection and alarm systems, including electronic supervision for other fire-detection and fire-suppression systems
- water fire-suppression systems
- emergency radio communication systems
- emergency lights
- emergency breathing-air systems and components

These preoperational testing requirements are consistent with the guidance in BTP CMEB 9.5-1, and are acceptable.

9.5.1.5 Administrative Controls

ABB-CE has indicated that the COL applicant will establish administrative controls to govern various details of operations of the plant in conformance with the guidelines of BTP CMEB 9.5-1. This is designated as DSER COL Action Item 9.5.1.5-1. This is acceptable.

The staff will perform a detailed review of the administrative controls during the plant-specific licensing process of an application referencing the ABB-CE System 80+ design. Items of interest under the administrative controls review will include:

- control of combustible materials such as combustible/flammable liquids and gases, fire-retardant-treated wood, plastic materials, and dry ion-exchange resins
- transient combustible materials and general house-keeping, including health physics materials
- open-flame and hot-work permits and cutting and welding operations
- quality assurance with respect to fire protection system(s) components, installation, maintenance, and operation

- qualification of fire-protection engineering personnel, fire-brigade members, and fire-protection system(s) maintenance and testing personnel
- instruction, training, and drills provided to fire-brigade members

9.5.1.6 Evaluation of Individual Fire Areas

As described in CESSAR-DC Section 9.5.1.12, the COL applicant must complete the fire hazards analysis using the Fire Hazards Assessment as the base document. The Fire Hazards Assessment will provide the programmatic framework for the plant fire protection program and the fire hazards analysis that will be developed. The fire hazards analysis will include but is not limited to the following:

- Consider the fire hazard for each room area or zone of the plant.
- Determine the consequence of fire in any location of the plant on the ability to safely shut down the reactor or on the ability to minimize and control the release of radioactivity to the environment.
- Specify measures for fire prevention, fire detection, and fire suppression.

ABB-CE has stated that the completion of the fire hazards analysis is a COL applicant item.

In the DSER, the staff stated that ABB-CE, in Section 6.0 of the Fire Hazards Assessment for the System 80+ design, stated its strategies for defining an acceptable level of risk. These strategies are presented in four steps.

- (1) Section 6.2, "Categories of Risk," defines four categories of risk.

Category 1: Any fire loss is unacceptable. No Category 1 fire areas are permitted in the System 80+ design.

Category 2: Some fire damage is acceptable, but loss of function is unacceptable.

Category 3: Loss of equipment function is acceptable, but fire propagation is unacceptable.

Category 4: Any possible fire loss is acceptable. Hazard or value is too low to justify expense of any automatic fire protection.

- (2) Section 6.3, "Fire Protection for Categories," specifies the level of fire protection to be provided for each of the four categories listed in Section 6.2.

Category 1: Such categories are not permitted.

Category 2: Automatic detection; fixed, automatic suppression systems.

Category 3: Automatic detection, manual fire suppression, automatic fire suppression if mandated by fire hazard analysis.

Category 4: Does not merit installation of detection and suppression system. Manual fire-fighting capability only.

- (3) Section 7.7, "Redundant Fire Areas Containing Safe Shutdown Equipment," lists each fire area containing safe-shutdown equipment, the safe-shutdown equipment contained in that area, and the redundant area.

- (4) ABB-CE describes in detail each of the 165 fire areas in the plant.

The staff reviewed about 50 of the approximately 165 detailed description of the fire area (several fire area numbers are reserved for future assignment). During the review, the staff noted a number of concerns that appeared to be common to most of the fire area descriptions. Since most of the staff concerns were repeated several times, they were listed as general concerns rather than fire area by fire area. Where appropriate, specific fire area references are given as examples.

- (1) Several fire areas designated as Category 2 have no automatic suppression provided, even though this is stated as a requirement in Section 6.3 (e.g., see fire areas 1, 2, 3, 4, 21, 22, and 24). This was designated as an DSER Open Item 9.5.1.6-1.

In a letter dated June 11, 1993 (LD-93-090), ABB-CE stated that the Fire Hazards Assessment described categories of risk, which are used to determine the type of fire protection features provided in each fire area. The Fire Hazards Assessment has been revised to describe Category 2 as those areas for which BTP CMEB 9.5-1 stipulates specific fire-protection features, including suppression systems, detection systems and fire barriers. Areas classified as Category 2 will be protected in accordance with the BTP CMEB 9.5-1. On this basis, DSER Open Item 9.5.1.6-1 is resolved.

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- (2) Some Category 2 areas state that automatic fire suppression is to be determined (TBD) (e.g., see fire areas 9, 10, 11, and 12). This was designated as DSER Open Item 9.5.1.6-2.

In a letter dated June 11, 1993 (LD-93-090), ABB-CE stated that the Fire Hazards Assessment describes categories of risk, which are used to determine the type of fire-protection features provided in each fire area. The Fire Hazards Assessment has been revised to describe Category 2 as those areas for which BTP CMEB 9.5-1 stipulates specific fire-protection features, including fire-suppression systems. Areas classified as Category 2 will be protected in accordance with the BTP CMEB 9.5-1. On this basis, DSER Open Item 9.5.1.6.2 is resolved.

- (3) All structural members (walls, floors, and ceilings), doors, dampers, and penetration seals are qualified by laboratory test or engineering analysis. The staff can see no instance for a new design where analysis would be acceptable in lieu of actual test. Should ABB-CE actually propose this for a specific plant, it must provide conclusive justification. This was designated as DSER Open Item 9.5.1.6-3.

In a letter dated June 11, 1993 (LD-93-090), ABB-CE stated that it is expected that qualification for the structural members (walls, floors, and ceilings), doors, dampers, and penetration seals will be verified through laboratory testing. However, in the course of the detail design, it may be necessary, for safety or security reasons, to utilize components that have not been tested for specific fire purposes. For example, the doors used for containment spray that have been tested in accordance with ASTM E-119 may not be available. In this case, an engineering analysis would be performed to assure that the entire door assembly meets or exceeds the same construction requirements required of doors that have been tested to ASTM E-119 standards. The analysis will be conservative and will assure that the appropriate level of fire protection is afforded the component.

If a deviation analysis is required for any component, it will be documented in the fire hazard analysis during detailed review design and reviewed by the NRC as part of the COL application. On this basis, DSER Open Item 9.5.1.6-3 is resolved.

- (4) Some Category 2 areas state that an engineering analysis will be performed to verify that automatic suppression is not needed (e.g., see fire areas 34

and 35). This was designated as DSER Open Item 9.5.1.6-4.

In a letter dated June 11, 1993 (LD-93-090), ABB-CE stated that engineering analysis will not be used to determine provisions of fire protection features for regulatory compliance.

The Fire Hazards Assessment describes categories of risk, which are used to determine the type of fire-protection features provided in each fire area. The Fire Hazards Assessment has been revised to describe Category 2 as those areas for which the BTP CMEB 9.5-1 stipulates specific fire-protection features, including suppression systems, detection systems, and fire barriers. Areas classified as Category 2 will be protected in accordance with BTP CMEB 9.5-1.

Engineering analysis will only be used to determine if it is beneficial to install fire protection, in addition to those features provided for regulatory compliance, to meet such other design objectives as personnel safety and unit availability.

The Fire Hazards Assessment has been revised to stipulate such specific fire-protection features such as suppression systems, detection systems, and fire barriers for areas as indicated by the BTP CMEB 9.5-1. ABB-CE commits to conform to BTP CMEB 9.5-1 (and SECY-90-016) for the installation of fire-protection features. On this basis, DSER Open Item 9.5.1.6-4 is resolved.

- (5) The normal HVAC system will be utilized for smoke removal from any area with a fire, and for smoke control to prevent migration from an area with a fire to other fire areas. In the DSER the staff stated that ABB-CE has not submitted details of operation of the HVAC system operating in the smoke-control/smoke-purge mode. Therefore, the staff is unable to complete its review of this mode of operation of the System 80+ HVAC system design. This was designated as DSER Open Item 9.5.1.2.2-2.

As previously discussed in Section 9.5.1.2.2 of this report, ABB-CE stated that the HVAC system is designed to remove smoke and mitigate smoke migration beyond the area of origin in the event of fire. The dedicated fans for smoke purge are designed to exhaust at a minimum of 945 L/minute/m² (3 cfm/ft²) of floor area. The normal ventilation is designed to provide an air flow of 315 L/minute/m² (1 cfm/ft²) of floor area or more.

In the Amendment U to CESSAR-DC Section 9.5.1.8.2, ABB-CE further indicated that the ventilation system will be designed in accordance with NFPA Standard 92B, "Guide for Smoke Management Systems in Malls, Atriums and Large Areas." ABB-CE's proposed HVAC design complies with the guidance of BTP CMEB 9.5-1 and SECY-90-016 and is acceptable. On this basis, DSER Open Item 9.5.1.2.2-2 is resolved.

- (6) ABB-CE must confirm that no penetrations exist in the three-hour-rated barriers separating fire areas containing redundant trains of safe shutdown equipment. This was designated as DSER Open Item 9.5.1.6-5.

In a letter dated June 11, 1993 (LD-93-090), ABB-CE stated that with few exceptions, there are no openings in the 3-hour-rated fire wall between redundant equipment required for safe shutdown. In cases where there are penetrations through the fire wall, they are provided with a rated assembly equivalent to the fire wall. The System 80+ design of maintaining a 3-hour barrier between redundant equipment is consistent with BTP CMEB 9.5-1 and SECY-90-016 and is acceptable. On this basis, DSER Open Item 9.5.1.6-5 is resolved.

9.5.1.7 Summary

In the DSER, the staff indicated that the Fire Hazards Assessment was still in preliminary form and was not sufficiently detailed to permit the staff to evaluate it completely. Subsequently, ABB-CE stated in the CESSAR-DC Section 9.5.1.12, that the licensee will use the Fire Hazards Assessment as the base document to develop a fire hazard analysis. The fire hazard analysis is considered a COL applicant item. In addition, the administrative controls will be the responsibility of the COL applicant (previously noted as DSER COL action item 9.5.1.5-1). Therefore, the staff will review the fire hazards analysis and administrative controls for each new plant.

The staff concludes that the fire protection program's design criteria and bases are acceptable and comply with the requirements of 10 CFR 50.48, GDC 3 and 5, and SECY-90-016. This conclusion is based on the System 80+ design meeting the guidelines of BTP CMEB 9.5-1 as well as the industry standards with respect of fire protection. In meeting these guidelines, ABB-CE has provided an acceptable basis for the design and location of safety-related structures and systems to minimize the probability and effect of fires and explosions; has used noncombustible and heat-resistant materials wherever practical; has

provided fire-detection and fire-fighting systems of appropriate capacity and capability to minimize adverse affects of fire on safety-related systems. In addition, the System 80+ design does not contain any shared structures, systems, or components.

9.5.2 Communications Systems

The staff reviewed the proposed design of the System 80+ communications systems to determine its conformance to the acceptance criteria in SRP Section 9.5.2 and to the guidance provided in the EPRI URD for ALWR. ABB-CE committed to adhere to the guidance in the EPRI URD.

The acceptance criteria in the SRP are based in part on the similarity of the communications systems design with that of previously reviewed plants with satisfactory operating experience. The SRP criterion for assessing system design capability is: The communication system is acceptable if its integrated design will provide effective communication between plant personnel in all vital areas during normal plant operation and during the full spectrum of accident or incident conditions (including fire) under the maximum potential noise levels. Furthermore, the scope of the evaluation should include a review of the design basis, design criteria, system description, and the supporting analyses that demonstrate the effectiveness of the system when maximum noise levels are being generated during accident or incident conditions to verify that the communication system will function effectively.

In Section 4.6.1 of Chapter 10 of the URD, EPRI stated that the scope of Section 4.6 includes both in-plant communications for operations and maintenance, and communications outside the plant, such as communication with a load dispatcher as well as with agencies for which communications is needed to support emergency operations.

In Section 4.6.2.2 of Chapter 10 of the URD, EPRI required the designer of the communication system to analyze the specific communication needs and the specific design requirements in the design-basis documentation, and to consider in this analysis specific communication tasks that must be performed by security personnel.

In Section 4.6.2 of Chapter 10 of the URD, EPRI stated that the primary, dedicated means of communication between operators during normal or emergency operation will be by portable, wireless communication equipment supported by appropriate base stations, antennae, amplifiers, and repeaters. A plantwide paging system and in-plant telephone system will be included. Offsite communications will be accomplished primarily with dedicated phone links. The staff notes that extensive wireless communication will increase the emphasis on consideration

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of electromagnetic interference (EMI) in the design of the man-machine interface system (M-MIS). The use of this type of communication system is not specifically excluded in NRC regulations; in fact, it may be necessary to use it in order to comply with the requirements of 10 CFR Part 73.

In Section 4.6.2 of Chapter 10 of the URD, EPRI also specified that there will be a frequency allocation plan to ensure that there is no interference between communication systems and to ensure compatibility with the EMI/RFI measures taken by the electronics and computer designers.

ABB-CE committed to install a radio system that conforms to the requirements of 10 CFR 73.55(f), and a crisis management radio system that conforms to NUREG-0654. Such commitment acceptable and where designated as DSER Confirmatory Item 9.5.2-1 COL Action Item 9.5.2-1. By SSAR Amendment L, ABB-CE added the following statement to CESSAR-DC Section 9.2.5.2.2.5: "In addition a security radio system is provided in accordance with 10 CFR 73.55(f) and a crisis management radio system provided in accordance with the intent of NUREG-0654." With the inclusion of this information, DSER Confirmatory Item 9.5.2-1 is resolved.

ABB-CE stated that the communication systems are designed to provide effective communications between all areas of the plant and plant site, including all vital areas of the plant. Additionally, ABB-CE stated that the communication systems are designed to provide an effective means to maintain contact with plant personnel and offsite utility and regulatory officials during normal conditions and such abnormal conditions as fire, accidents, and plant testing. ABB-CE did not submit the physical location of the communications devices; however, ABB-CE committed to install the equipment in all vital areas of the plant. Specific location of equipment will be determined on a site-specific basis. ABB-CE's commitments are acceptable. This is designated as DSER COL Action Item 9.5.2-2.

The communications systems consist of the intraplant private automatic business exchange (PABX) telephone system, the intraplant public address (PA) system, the intraplant sound-powered telephone systems, and offsite communications via public telephone lines and the utility's private network, which is connected to the PABX telephone switch.

The PABX is independent of the intraplant PA system and the intraplant sound-powered telephone system. The staff evaluated the inspection, test, analysis and acceptance criterion (ITAAC) to verify the independence of these

systems. This ITAAC was found to be acceptable; therefore, DSER Open Item 9.5.2-1 is resolved.

EPRI stated that potential high-noise areas will be considered in the design of the communication system design. The plant designer must provide adequate means to alert personnel in high-noise environments that they are to use the communication systems. The COL applicant will install noise-canceling devices or sound isolation booths or both in noisy areas. Additionally, the communications system includes a PA system. This is COL Action Item 9.5.2-3.

ABB-CE stated that the System 80+ design employs a portable, wireless communication system to provide communication capability among control room operators, equipment operators, and maintenance technicians for routine and emergency operations. In addition to portable and wireless transmitters/receivers, this system is supported by appropriate base stations, antennae, amplifiers and/or repeaters. Specialized system portable and wireless transmitters/receivers are provided for use in operations requiring respirators or underwater diving equipment. This is acceptable to the staff and on this basis, DSER Open Item 9.5.2-2 is resolved.

In Section 4.6.3 of Chapter 10 of the URD, EPRI required that there will be no interference between the communications systems and M-MIS equipment. The communication system designers are required to define the worst-case electromagnetic emissions from the communications equipment, including the type, magnitude, frequency content, and locations. All potential uses of the communications equipment will be considered, including maintenance activities. The communications system will also be protected from the M-MIS equipment to the extent that clear communications will be maintained.

ABB-CE stated that all communications systems are inspected, checked, and tested for operability after installation to ensure proper operation and coverage. Additionally, ABB-CE stated that the normal and continued use of the systems provides the basis for inspections of the systems. The commitment to inspect all communications systems after installation to ensure their proper functioning is acceptable.

Security Considerations

In CESSAR-DC Section 9.5.2.2, ABB-CE states that intraplant communication systems include PABX telephone, PA, and sound-powered telephone systems. ABB-CE also states that offsite communications includes a security radio system in accordance with 10 CFR 73.55(f).

In addition to requiring communication with offsite law enforcement authorities, in 10 CFR 73.55(f), the staff requires intraplant security communications. The intraplant security communications requirement (10 CFR 73.55(f)(1)) specifies that plant security officers must be capable of continuous communication with the central and secondary alarm stations. Security response forces of operating reactors rely upon individual mobile radios for communications during tactical movements. In CESSAR-DC Section 9.5.2, ABB-CE provides only for telephones for intraplant communications. In CESSAR-DC Section 9.5.2.2.4, ABB-CE addresses a security radio system for offsite communications, but not for onsite communications with security officers. In Item 18 of ABB-CE's December 21, 1990, letter on differences between the URD and the System 80+ standard design, ABB-CE stated that wireless communications can be used where needed, but that sound-powered phones avoid such problems as interference with control systems and continuity of coverage. However, in Section 4.6.3 of URD Chapter 10, EPRI specified that the plant designer should ensure that effective communication is not impeded by problems of transmission through thick concrete walls, high-noise areas, personnel protective equipment, too few communication lines, or interference with or from other electronic or electrical equipment. In RAI Q500.23, the staff asked ABB-CE to detail consideration given to assuring the capability for security officers to be in continuous intraplant communication at System 80+ standard plants. In a response of December 17, 1991, ABB-CE stated that plant security is expected to depend on wireless intraplant communications, but that the plant owner/operator will choose the system. The staff did not consider this response adequate since ABB-CE did not address design provisions that may be necessary to allow this capability, such as shielding of instrument transmitters and logic cabinets from radio-frequency interference (RFI), fiber-optic cabling, and radio repeaters within buildings, and did not give reasons for deviating from the EPRI ALWR URD. In NRC Information Notice 83-83, "Use of Portable Radio Transmitters Inside Nuclear Power Plants," the staff stated: "As newer plants are built that use more solid-state equipment . . . more cases of RFI by portable radio transmitters are likely to result If plant operations make the use of portable radio transmitters near RFI-sensitive equipment either necessary or likely in an emergency, then administrative prohibitions are not adequate and the licensee should consider hardware fixes." This was designated as DSER Open Item 9.5.2-3.

Subsequently, in Amendment J to CESSAR-DC, ABB-CE added the following statement in Section 13.6: "The security communications subsystem shall meet the following requirements: . . . Each on-site security officer, watchman, or armed response individual shall be provided

with continuous communications with an individual in each continuously manned alarm station. This may be accomplished by using multi-frequency radio or microwave-transmitted two-way voice communications." Requiring continuous wireless communication between security officers, ensures adequate communication for the security organization. This additional information is acceptable to the staff. On this basis, DSER Open Item 9.5.2-3 is resolved.

9.5.3 Lighting System

The normal lighting system supplies normal illumination under all plant operating, maintenance, and test conditions. In CESSAR-DC, Table 9.5.3-1, ABB-CE summarizes typical illuminance ranges for normal lighting. The lighting fixtures are designed and located so that plant personnel can maintain and replace lights effectively and safely. ABB-CE indicates that the circuits to the individual lighting fixtures are staggered as much as possible and that separate electrical circuits feed these staggered circuits to ensure that some lighting is retained in the event of a circuit failure. The failure or unavailability of a single lighting transformer does not affect the ability of the system to operate normally. The normal lighting system is part of the plant's permanent non-safety system. Therefore, the normal lighting system has power as long as power from an offsite power source or a standby non-safety source (combustion turbine generator (CTG)) is available.

In CESSAR-DC, ABB-CE indicated that the emergency lighting system for System 80+ design is accomplished by two systems: conventional ac fixtures fed from Class 1E power sources and Class 1E dc self contained, battery operated lighting units. This emergency lighting is located in vital areas throughout the plant. These areas are determined by performing hazard analyses and establishing plant emergency procedures. ABB-CE indicates that the operator will need to gain access to several vital areas, i.e., the main control room (MCR), the technical support center, the operations support center, the remote shutdown panel room, the sample room, the hydrogen recombiner rooms, and stairwells and passageways. ABB-CE states that as it completes hazard analyses and plant emergency procedures, it is designating other areas as vital, such as the EDG rooms, the steam-driven emergency feedwater pump rooms, and the pathways from the control room to these rooms.

The security lighting system is considered part of the permanent non-safety system and is fed from the AAC Source (Combustion Turbine). Selected portions of the security lighting system essential to maintaining adequate

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plant protection are powered from a non-Class 1E battery power from a non-Class 1E battery power source.

ABB-CE submitted information on the design of the normal, emergency, self-contained dc lighting units and security lighting to demonstrate that the lighting in the normal and vital plant areas as well as passageways to and from these areas is adequate. After installation, each lighting system will be inspected, checked, and tested to verify that it is operable and provides proper coverage.

In the DSER, the staff asked ABB-CE to address the following staff concerns and EPRI guidelines:

- (1) The staff will verify that the completion of the hazards analyses and plant emergency procedures are included as commitments and that appropriate inspections, tests, and/or analyses are included as part of ITAAC to verify implementation of their design commitments.
- (2) State what method will be used to distinguish between the normal, emergency, and security lighting cables and circuits to ensure they are physically identified and separated.
- (3) Confirm that the Class 1E distribution system supplies at least one of the circuits supplying lighting fixtures for the normal lighting system in safety-related areas (other than main control room) and in access routes to these areas. The other lighting circuit can be supplied from a non-Class 1E electrical division backed-up by the CTG.
- (4) Integrate the emergency lighting system in the main control room with the normal lighting system and design it so that alternate emergency lighting fixtures are fed from separate safety divisions.
- (5) Design the emergency lighting installations that serve the main control room and those other areas of the plant where safe shutdown operation may be performed so that these installations will continue to function during and after a DBE.
- (6) Confirm which part of the emergency lighting system will not be qualified as Class 1E.

The staff stated in the DSER that it would verify that these six aspects are included in the design commitments. This was designated as DSER Open Item 9.5.3-1.

In Amendment T, to CESSAR-DC, ABB-CE supplemented the information on lighting systems as follows:

ABB-CE has designated that the following areas of the plant be included as vital areas; the main control room, the technical support center, the operations support center, the remote shutdown panel room, the sample room, the hydrogen recombiner rooms, electrical system areas, main steam valve houses, chemistry labs, the EDG rooms, the stairway which provides access from the control room to the remote shutdown panel room, and other areas where operator access is required.

ABB-CE stated that associated emergency lighting will be located in vital areas of the plant. The associated emergency lighting system is used to provide acceptable levels of illumination throughout the station and particularly in vital areas in which emergency operations are performed upon loss of the normal lighting system. The associated emergency lighting system provides a minimum illumination level of 10 foot-candles in areas of the plant in which emergency operations are performed. For other areas of the plant covered by the emergency lighting system, a minimum illumination level of 2 foot-candles is provided. This addresses Item (1) and is acceptable because it ensures that the illumination levels are in accordance with the guidelines of the Illuminating Engineering Society (IES) Lighting Handbook, as recommended by the IES.

ABB-CE stated that the criteria for the physical identification of lighting cables and circuits are consistent with the criteria for physical identification and separation of Class 1E and non-Class 1E cables and circuits as described in IEEE-384 and RG 1.75, which are part of CESSAR-DC, Chapter 8, "Electrical Power Systems." This addresses Item (2) and is acceptable because it ensures that the cables and circuits are physically identified and separated.

The normal lighting system provides general illumination throughout the plant. The circuits to the individual lighting fixtures are staggered as much as possible with the staggered circuits fed from separate electrical divisions to ensure some lighting is retained in the room in the event of a circuit failure. The normal lighting system is considered part of the plant permanent non-safety systems. As such, the normal lighting system is energized as long as power from an offsite power source or a standby non-safety source (Combustion Turbine) is available. The Combustion Turbine is designed to start automatically within two minutes from the onset of a LOOP event. ABB-CE stated that all lighting fixtures and other components of the lighting system located in normally occupied areas or in areas containing safety equipment are supported so as to enhance the earthquake survivability of these components and to ensure that they do not present a personnel or equipment hazard when subjected to a seismic loading of a DBE. This addresses Item (3) and is acceptable because it ensures that the normal lighting system is energized as

long as power from an offsite power source or a standby non-safety source (CTG) is available under normal and abnormal plant conditions.

ABB-CE indicated that the emergency lighting system in the main control room will be integrated with the normal lighting system. The emergency lighting in the MCR will be configured so that at least two circuits of lighting fixtures are powered from different Class 1E divisions to ensure lighting is retained in the event of a circuit failure. This addresses Item (4) and is acceptable because it minimizes the effects of a single circuit failure during normal plant operations.

ABB-CE indicated that the associated emergency lighting system in the main control room will maintain minimum illumination levels in the main control room during all operating and emergency conditions, including a station blackout (SBO). The Class 1E and associated emergency lighting installations which serve the main control room and other areas of the plant where safe shutdown operations may be performed are designed to remain functional during and after a DBE. This addresses Item (5) and is acceptable because it ensures the associated emergency lighting in the main control room and other vital areas of the plant will remain functional during normal and abnormal plant conditions.

ABB-CE indicated that the Emergency lighting is composed of two systems, (1) conventional ac fixtures fed from Class 1E ac power sources which, excluding the fixtures, are qualified as associated circuits and, (2) dc self contained battery-operated lighting units which are qualified as Class 1E circuits. This addresses Item (6) and is acceptable because it helps ensure adequate lighting is maintained.

ABB-CE indicated that the lighting system for the System 80+ is designed to provide illumination throughout the plant and plant site. In accordance with SRP Section 9.5.3, lighting levels and illuminance ranges for the System 80+ design comply with the IES Lighting Handbooks recommended intensities.

ABB-CE indicated that the luminaries are of a proven design with long life and low maintenance requirements, such as fluorescent, metal-halide, and high pressure sodium lamps. Mercury vapor lamps are not used in fuel-handling areas. Fluorescent luminaries are normally used in plant stairs and stairwells; around switchgear, motor control centers, and instrumentation racks; and to supplement high intensity discharge (HID) luminaries.

ABB-CE stated that the Class 1E dc self contained battery-operated lighting units are designed for eight hours of

continuous operation following loss of normal power. Each Class 1E unit has a self-contained battery pack unit containing a rechargeable battery with a minimum eight-hour capacity. The Class 1E lighting units are supplied AC power from the same power source as the normal lighting system in the area in which they are located. The loading of these Class 1E lighting units is not to be greater than 80 percent of the rated capacity with additional derating for temperature variations, where appropriate. The bulbs are positioned so that adequate illumination is provided and is unobstructed by plant equipment and components. The Class 1E units also contain a time delay so that the lights turn off on the resumption of power only after an adequate delay to allow normal lighting to restart.

ABB-CE stated that additional non-Class 1E dc self-contained battery-operated lighting units are installed throughout the plant to provide emergency lighting for personnel safety in accordance with the applicable sections of the National Electric Code and the Life Safety Code of the National Fire Protection Association.

On this basis, the staff concludes that the lighting system for the ABB-CE System 80+ design conforms to the guidance in SRP Section 9.5.3 and the IES lighting handbooks and is acceptable. On this basis, DSER Open Item 9.5.3-1 is resolved.

Security Considerations

In CESSAR-DC Section 9.5.3.1, ABB-CE states that the security lighting system provides illumination required to monitor the isolation zones and all outdoor areas within the plant protected perimeter and complies with the intent of NUREG/CR-1327. By means of Amendment E, ABB-CE stated in CESSAR-DC Section 9.5.3.2.2 that the security lighting system is part of the permanent non-safety systems loads and is fed from an uninterruptible power supply (UPS) connected to a non-safety battery. In a December 17, 1991, response to RAI Q500.20, ABB-CE proposed to delete the reference to an UPS and change Section 9.5.3.2.2 to read instead: "The security lighting system is considered part of the permanent non-safety systems and is fed from the AAC source (combustion turbine), which is located in a secure vital area for protection." However, as described in CESSAR-DC Section 8.3.1.1.5.1, the AAC could take 10 minutes to start (from the onset of an LOOP event), and additional time for load sequencing before security lighting would be restored. The staff concluded that the proposed change is inconsistent with CESSAR-DC Section 9.5.3.1 and did not conform with the URD requirement for uninterruptible power for those portions of the security lighting that are essential to plant protection following interruption of normal power. Inconsistency between CESSAR-DC Section 9.5.3.1 and

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the changes proposed for CESSAR-DC Section 9.5.3.2.2, and ABB-CE's reason for deviating from the URD requirements for uninterruptible power for those portions of the security lighting that are essential to plant protection, was designated as DSER Open Item 9.5.3-2. By CESSAR-DC Amendment J, ABB-CE stated in CESSAR-DC Section 8.3.1.1.5.1, that the AAC is designed to start automatically within two minutes from the onset of an LOOP event. As stated in CESSAR-DC Section 9.5.3.2.2, the security lighting is part of the permanent non-safety systems which is fed from the AAC. Selected portions of the security lighting system essential to maintaining adequate plant protection are powered from an uninterruptible power supply. Requiring uninterruptible power for those portions of the security lighting that are essential to plant protection is consistent with the URD requirements. This additional information has also removed the inconsistency between CESSAR-DC Section 9.5.3.1 and the changes proposed for CESSAR-DC Section 9.5.3.2.2. On this basis, DSER Open Item 9.5.3-2 is resolved.

In CESSAR-DC Section 9.5.3.2.2, ABB-CE states that the security lighting system provides a minimum illumination of 0.2 foot-candle at ground level. In CESSAR-DC Table 9.5.3-1, ABB-CE lists 2 to 5 foot-candles as typical illuminance ranges for normal exterior area lighting. Those illumination levels give a range of ratios of typical to minimum illumination of 10:1 to 25:1. However, in NUREG/CR-1327, the NRC contractor recommends that for security lighting of the protected area's isolation zone, the average to minimum illumination ratio should not exceed 4:1, with less than 3:1 preferred. In a letter of December 17, 1991, responding to RAI Q500.27, ABB-CE proposed to add the following to Section 9.5.3.2.2: "The Owner-Operator shall provide a security lighting system that will meet closed-circuit television (CCTV) illumination requirements within camera viewing areas to permit prompt assessment of intrusion alarms." This was designated as DSER Confirmatory Item 9.5.3-1. By CESSAR-DC Amendment J, ABB-CE incorporated the above statement to CESSAR-DC Section 9.2.3.2.2, resolving the staff's concern about adequate illumination for CCTV and resolving DSER Confirmatory Item 9.5.3-1.

9.5.4 Diesel Generator Engine Fuel Oil System

Section 9.5.4.1 of this report addresses compliance of all the diesel generator auxiliary systems with the requirements of GDC 2, 4, and 5. Section 9.5.4.2 of this report addresses issues specific to the diesel generator engine fuel oil system.

9.5.4.1 Diesel Generator Engine Auxiliary Support Systems (General)

There are two standby EDGs in the CESSAR-DC design. Each DG engine has the following auxiliary support systems which are addressed in detail in the CESSAR-DC sections indicated:

1. Diesel Generator Engine Fuel Oil System (Section 9.5.4.2),
2. Diesel Generator Engine Cooling Water System (Section 9.5.5),
3. Diesel Generator Engine Starting Air System (Section 9.5.6),
4. Diesel Generator Engine Lube Oil System (Section 9.5.7), and
5. Diesel Generator Engine Air Intake and Exhaust System (Section 9.5.8).

Adequacy of the systems is based on compliance with the requirements of GDC 2 for protection against natural phenomena, GDC 4 for protection from environmental and dynamic effects of equipment failure, and GDC 5 for sharing of SSCs between units, as well as the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." Compliance with the requirements of other GDC will be reviewed on a system-specific basis in later sections of this report.

The diesel engine vendor has not been selected. Therefore, ABB-CE has not fully defined the auxiliary support systems. Many component variables (such as tank size, pump capacity and, system flow rates) are dependent on the specific needs of the selected DG and particular site. However, the requirements and design characteristics specified, and committed to, for the DG auxiliary support systems in the CESSAR-DC offer reasonable confidence that NRC regulations will be satisfied.

All components and piping of the DG auxiliary support systems, except for a portion of the fuel oil storage and transfer system piping from the fuel oil storage tanks to the day tank and the lube oil system's clean and used lube oil storage tanks and piping, are housed in a seismic Category I structure (DG building, diesel fuel storage structure) which provides protection from the effects of tornados, tornado missiles, turbine missiles, and floods. The portions of the fuel oil storage and transfer system's day tank vent and storage tank vents and fill connections which are exposed to outdoors are constructed with heavy-gauge pipe to provide protection from tornado missiles and are

located above the probable maximum flood level to provide protection from floods. Each fill connection has a locking dust cap and each vent line is turned down. The storage tanks can be filled and vented through the manway should fill or vent lines become impaired. Therefore, this complies with Position C.1 of RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," and Appendix Position 13 of RG 1.117, "Tornado Design Classification." On this basis, the staff concludes that the system complies with the requirements of GDC 2 with respect to structures housing the systems and the systems themselves being capable of withstanding the effects of externally and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the System 80+ is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The evaluation of the design of the DG auxiliary support systems with respect to the effects of postulated pipe failures is addressed in Section 3.6 of this report. The adequacy of the fire-protection system and the associated auxiliary support systems is addressed in Section 9.5.1 of this report.

The staff reviewed the designs of the DG auxiliary support systems with respect to the recommendations of NUREG/CR-0660. NUREG/CR-0660 made specific recommendations on increasing the reliability of emergency DG at nuclear plants. Table 9.1 of this report summarizes compliance of the System 80+ DG auxiliary support systems with the recommendations of NUREG/CR-0660.

NUREG/CR-0660 recommends that, as a minimum, operations and maintenance personnel receive training that is provided by the vendor, or that is equivalent to vendor training, since a lack of proper training has contributed to the degradation of DG reliability. Increased attention to investigative testing, replacements and adjustments as part of preventive maintenance, and increased root-cause analysis of system (component) failures are site specific and will need to be addressed as part of the site-specific application. This was designated as DSER COL Action Item 9.5.4.1-1. Licensing Document LD-92-006, dated January 24, 1992, contains a note that ABB-CE, in response to RAI Q410.123.3, will revise CESSAR-DC Section 8.3.1.1.4 to include operational requirements for training of personnel and the establishment of a preventive maintenance program. In response to NRC's ITAAC Comment 2.7.19-4d, ABB-CE committed to add Section 8.3.1.1.4.13 to the next CESSAR-DC amendment to incorporate the RAI Q410.123.3 response. In

CESSAR-DC Amendment T, ABB-CE added Section 8.3.1.1.4.13 to the CESSAR-DC which includes a statement about the operational requirements for training of personnel and the establishment of a preventive maintenance program. The staff finds the addition of this section in the CESSAR-DC acceptable because it complies with the recommendation in NUREG/CR-0660. Therefore, DSER COL Action Item 9.5.4.1-1 is acceptable.

Criteria for the test loading of the DG and loading of the generator to 25 percent for one hour after up to eight hours of no-load or light-load operation, have been included as a combined license action requirement. DGs are to be tested in accordance with RG 1.108 requirements, and other support system tests are to be performed periodically. For all DG auxiliary support systems, the test and calibration frequencies will be evaluated on a plant-specific basis. This was designated as DSER COL Action Item 9.5.4.1-2. ABB-CE has included this information in Sections 9.5.4, 9.5.5, 9.5.6, 9.5.7, and 9.5.8 of Amendment Q to the CESSAR-DC by stating that the COL applicant will make testing and calibration frequencies for each DG support system available for NRC review. In addition, ABB-CE stated that each DG support system will be tested in accordance with the TSs. The staff finds this acceptable. On the basis of this evaluation, DSER COL Action Item 9.5.4.1-2 is acceptable.

NUREG/CR-0660 also recommends that provisions to reduce the effect of dust and dirt on DG operation and reliability be included as requirements. LD-92-006, dated January 24, 1992, contains an addition to CESSAR-DC Section 9.4.4.3 stating that: "The fresh air intake on the Diesel-Generator Building Ventilation System is located 20 feet above grade to minimize the intake of dust into the building. All exposed diesel-generator building interior surfaces are to be painted to aid in dust control." In accordance with SRP Section 9.4.5.III.5, this describes an acceptable alternative to utilizing dust-tight sealed cabinets.

In CESSAR-DC Section 8.3.1.1.4, ABB-CE addresses the vibration concerns expressed in NUREG/CR-0660. The emergency DG controls and monitoring instrumentation, with the exception of the sensors and other equipment that must necessarily be mounted on the DG or its associated piping, are installed in free-standing, floor-mounted panels that are designed for their normal vibration environment and qualified to seismic Category I requirements.

On the basis of the above review, the staff concludes that all of the DG auxiliary support systems comply with the requirements of GDC 2, 4, and 5 with respect to protection against natural phenomena, protection from environmental and dynamic effects of equipment failure, and

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Table 9.1 Conformance to NUREG/CR-0660 recommendations for diesel generator auxiliary support systems

Recommendation	Conformance	FSER Section
(1) Moisture in air starting system	Yes	9.5.6
(2) Dust and dirt in diesel generator room	Yes	9.5.4.2
(3) N/A to support systems	--	--
(4) Personnel training	Specific site	9.5.4.1
(5) Automatic prelube	Yes	9.5.7
(6) Testing, test loading, and preventive maintenance	Specific site	9.5.4.1
(7) Improve the identification of cause of failures	Specific site	9.5.4.1
(8) Diesel generator ventilation and combustion air systems	Yes	9.5.8
(9) Fuel storage and handling	Yes	9.5.4.2
(10) High-temperature insulation	-- ¹	9.5.4.1
(11) Engine cooling water	Yes	9.5.5
(12) Concrete dust control	Yes	9.5.4.1
(13) Vibration of instruments	Yes	9.5.4.1

¹ The staff considers explicit conformance unnecessary in view of the equivalent reliability provided by the design, margin, and qualification testing requirements that are normally applied to emergency standby diesel generators.

sharing of SSCs between units, as well as the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," and are, therefore, acceptable.

Security Considerations

The staff considers that the DG and its support systems (fuel, cooling water, starting air, lube oil, exhaust, field flashing, and instrumentation and controls) are vital systems; therefore, as required by 10 CFR 73.55(c),

access to all DG and vital support systems' components should require passage through two barriers. (Locked security doors controlling access between two adjacent vital areas are not desired, if access to each vital area is otherwise controlled.) The description in CESSAR-DC Sections 9.5.5, 9.5.6, 9.5.7, and 9.5.8 of the protected location of DG components (i.e., the DG building) did not address protection from sabotage. The DG building was not included as a vital area in ABB-CE's response of September 28, 1989, to RAI Q500.7; also, ABB-CE's response to followup RAI Q500.21 referred to the DG

building as a vital area only in the sense of radiation protection guidance of RG 1.97 and NUREG-0737. Vital designation of the DG system in the sense of 10 CFR Part 73 was designated as DSER Open Item 9.5.4.1-1. Subsequently, ABB-CE revised CESSAR-DC Sections 9.5.4, 9.5.5, 9.5.6, 9.5.7, 9.5.8 and 9.5.9 to designate the DG's systems as vital systems. This change resolved the staff's concerns about designation of the DG systems and DSER Open Item 9.5.4.1-1 is resolved.

In a letter of February 28, 1992, EPRI advised ABB-CE to add Section 9.5.1.4.2 to the CESSAR-DC stating that the diesel fuel storage structure is a seismic Category I structure within the scope of the COL applicant and requiring ABB-CE to build a structure that is in the "vital protection area" and that will "withstand the effects" of a sabotage event. The staff interprets that terminology to mean that access to equipment within the diesel fuel storage structure requires passage through both the protected area fence and an additional vital barrier. The proposed vital area designation complies with the NRC Review Guideline 17 criterion that seismic Category I equipment be considered vital equipment and is acceptable. However, the proposed description of the ventilation system for the diesel fuel storage structure did not make reference to ventilation barrier guidance of RG 5.65 and was not sufficient for the staff to determine that a structure designed to these requirements will provide adequate vital barrier protection. Security barriers in the diesel fuel storage structure ventilation system was an issue to be resolved during review of the COL application. This was designated as COL Action Item 9.5.4.1-2. In CESSAR-DC Amendment Q, ABB-CE revised which support structures are in the scope of the CESSAR-DC System 80+ design. One of the additional structures included in the scope was the diesel fuel storage structure. The design of this structure does not need or contain a ventilation system. This design eliminates the need for the COL applicant to address security barriers for a diesel fuel storage structure ventilation system. The inclusion of this information into the CESSAR-DC eliminates the need for COL Item 9.5.4.1-2.

9.5.4.2 Diesel Generator Engine Fuel Oil System

The staff reviewed the DG engine fuel oil system (DGEFOS) in accordance with the guidance of SRP Section 9.5.4. DGEFOS provides for the storage of a seven-day supply of fuel oil for each DG engine, as necessary, to drive the EDG. Staff acceptance is based on complying with the requirements of GDC 2 for protection against natural phenomena, GDC 4 for protection from environmental and dynamic effects of equipment failure, GDC 5 for sharing of SSCs between units, and GDC 17 for availability of electric power systems. Compliance

with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report.

The staff based its review on the descriptive information and system diagrams in the CESSAR-DC (primarily Chapters 8 and 9) and the information supplied by ABB-CE in response to the staff's RAI.

There are two standby EDGs in the CESSAR-DC design. Each DG has a separate and independent DGEFOS division. Each DGEFOS division consists of two fuel oil storage tanks, with a combined capacity sufficient to supply fuel oil to power the diesel for seven days at full load, a fuel oil transfer valve, a day tank, an engine-driven fuel oil pump, and a motor-driven fuel oil booster pump. Each DGEFOS division is independent and physically separated from the other division serving the redundant DG. Thus, a single failure in one of the two divisions will affect only the associated DG.

Since there is no fuel oil transfer pump, fuel oil is transferred by gravity from the fuel oil storage tanks to the day tank. Level switches located in the day tank control the position of the fuel oil transfer valve. In the event the transfer valve remains in the closed position, the day tank low level alarm, indicating that 60 minutes of fuel remains at full load, allows the operator to manually open a valve on the bypass line to fill the day tank. In the event the transfer valve remains in the open position, fuel oil would continue to flow from the storage tanks to the day tank until the system reached hydrostatic equilibrium. Since there is no day tank overflow line, fuel oil would rise in the day tank vent pipe to a level that is equivalent to level of the fuel oil in the storage tanks, but well below the top of the vent.

All piping and components are located in seismic Category I buildings, except for a portion of the ANSI Class 3 seismically qualified piping from the fuel oil storage tanks to the day tank. The external surfaces of carbon steel tanks and underground components are coated for corrosion protection. In addition to being coated, the external surfaces of buried piping and tanks are protected from corrosion by an impressed-current cathodic protection system in accordance with NACE Standard RP-01-69, or other methods deemed appropriate on the basis of site-specific conditions. The diesel engine and engine-mounted components are constructed in accordance with IEEE Standard 387, "IEEE Standard Criteria For Diesel Generator Units Applied As Standby Power Supplies For Nuclear Power Generating Stations." The DGEFOS is designed and constructed in compliance with ANSI Standard N195, "Fuel Oil Systems for Standby Diesel Generators," except in regard to (1) the flame arresters on the storage tanks

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(these are not required specifically by SRP Section 9.5.1), (2) an overflow line from the day tank, and (3) the fuel oil transfer pumps. The DGEFOS fuel oil sampling and impressed current cathodic protection system surveillances are in compliance with the guidance of RG 1.137. The DGEFOS complies with the guidelines of RG 1.9 as it relates to system protective interlocks. All trips associated with the DGEFOS are bypassed during LOCA conditions. On this basis, the staff concludes that the DGEFOS complies with the requirements of GDC 17 with respect to the capability of the system to meet independence and redundancy criteria.

The System 80+ design utilizes two half-capacity fuel oil storage tanks per diesel; these provide the ability to operate the diesel off one tank while isolating and filling the other tank. After filling a storage tank, a period of not less than 24 hours is required to allow any stirred sediment to settle before aligning that tank to its respective diesel. The filtering and recirculation process is performed on a tank-by-tank basis, with the frequency of operation dependent on the results of a fuel oil inspection program. Accumulated water in the fuel oil storage tanks will be removed by the recirculation system through a sample connection provided on the recirculation pump discharge. Fuel oil stored in the day tank is periodically checked for water content. Accumulated water is drained out through the drain connection on the bottom of the tank.

The DGEFOS consists of the following instrumentation: fuel oil storage tank low- and high-level annunciators, low-low level alarm, 0-100 percent level indication, and the capability to use a stick gauge to measure fuel oil level; fuel oil recirculation filter inlet and outlet pressure indication; fuel oil day tank high- and low-level alarm, level indication, and fuel oil transfer valve control; fuel oil strainers (engine-driven pump and motor-driven booster pump) inlet and outlet pressure indication, and high differential pressure alarm; fuel oil filter high differential pressure alarm, differential pressure indication, outlet pressure indication, and low fuel oil pressure alarm; day tank retaining wall high- and low-level drain valve and lube oil transfer pump control, and high-high level alarm. The fuel oil transfer valve opens on day tank low level and closes on day tank high level. The day tank low level indicates that there is enough fuel oil remaining in the tank to run the DG at full load for 60 minutes. There are no temperature indications or controls indicated for the DGEFOS. The staff addressed this condition in an RAI. ABB-CE responded that all fuel oil equipment is located in heated buildings and that piping is installed below the frost line. ABB-CE stated that it would revise CESSAR-DC Section 9.5.4.3 to provide assurance that fuel oil temperature would remain above the cloud point. This was designated as DSER Confirmatory Item 9.5.4.2-1.

ABB-CE included this information in Section 9.5.4.2-1 of Amendment J to the CESSAR-DC. The staff finds this acceptable because it assures the fuel oil temperature will remain above the cloud point. Therefore, DSER Confirmatory Item 9.5.4.2-1 is resolved.

Specific design criteria, such as specific tank capacities, pump characteristics, and system flow parameters, were not given in the design description. The selection of these criteria will be included with each application and evaluated on a plant-specific basis. This was designated as DSER COL Action Item 9.5.4.2-1. ABB-CE has included a sentence in Section 9.5.4.2.1 of Amendment Q to the CESSAR-DC that states that the COL applicant will submit the specific DGEFOS design criteria. In addition, ABB-CE has included information on tank capacities in Section 9.5.4.2.1 of Amendment N to the CESSAR-DC. The staff finds this acceptable. On this basis, DSER COL Action Item 9.5.4.2-1 is acceptable. On the basis of the above review, the staff concludes that the design of the DGEFOS complies with the requirements of GDC 17 with respect to availability of electric power systems. As discussed in Section 9.5.4.1 of this report, the system also complies with the requirements of GDC 2, 4, and 5, as well as the recommendations of NUREG/CR-0660. Therefore, the staff concludes that the DGEFOS complies with the guidelines of SRP Section 9.5.4 and is acceptable.

9.5.5 Diesel Generator Engine Cooling Water System

The staff reviewed the DG engine cooling water system (DGECWS) in accordance with the guidance of SRP Section 9.5.5. The DGECWS maintains the temperature of the diesel engine within a safe operating range under all load conditions and maintains the engine coolant preheated during standby conditions. Staff acceptance is based on complying with the requirements of GDC 2 for protection against natural phenomena, GDC 4 for protection from environmental and dynamic effects of equipment failure, GDC 5 for sharing of SSCs between units, GDC 17 for availability of electric power systems, GDC 44 for provision for cooling systems, GDC 45 for inspection of cooling systems, and GDC 46 for testing of cooling systems. Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report.

The staff based its review on the descriptive information and system diagrams in the CESSAR-DC (primarily Chapters 8 and 9) and the information supplied by ABB-CE in response to the staff's RAI.

There are two standby EDGs in the CESSAR-DC design. An independent and physically separate closed-loop

DGECWS division is provided for each DG engine, receiving makeup water from the demineralized water system and rejecting heat to the CCWS. Thus, a single failure in one of the two divisions will affect only the associated DG. The jacket water standpipe serves as a surge tank and provides positive suction pressure for the circulation pump and for the keep-warm pump. The keep-warm pump, which is electric-motor-driven, operates continuously during engine standby to ensure that the system is completely filled with water. When the diesel starts, the engine-driven circulation pump operates to circulate cooling water through the system. The cooling water passes from the circulation pump through an Amot type or equivalent three-way thermostatic control valve which regulates the flow of water through the shell side of the jacket water cooler by diverting varying amounts through a bypass line. From the jacket water cooler, the cooling water flows through the tube side of either the lube oil cooler or the combustion air aftercoolers and then through the engine itself, returning to the standpipe. A small fraction of the flow from the discharge side of the combustion air aftercooler is diverted to the engine governor lube oil cooler.

In order to keep the engine warm during standby, water circulated by the keep-warm pump passes over a set of thermostatically controlled electric heating elements before leaving the standpipe. Cooling water from the keep-warm pump bypasses the jacket water cooler and thermostatic valve and thereafter follows the circulation pump flow path. If the switch is in the automatic position, power to the heater and keep-warm pump is automatically shut off when the diesel starts and has attained rated speed and is automatically started when the diesel is shut down.

Specific information regarding standpipe capacity, circulation and keep-warm pump characteristics, system volume, flow rates, and heat removal capabilities, plus expected water loss over a 7-day period, are designated as interface issues to be submitted once the DG vendor has been selected and diesel engine data are available. This information must be submitted as part of the application for a combined license. This was designated as DSER COL Action Item 9.5.5-1. ABB-CE addressed this issue in Section 9.5.5.2 of Amendment Q to the CESSAR-DC by stating the COL applicant will submit plant-specific information for the DGECWS. The staff finds this acceptable. Therefore, DSER COL Action Item 9.5.5-1 is acceptable.

The DGECWS is an ANSI Class 3 piping system. All components and piping are located within a seismic Category I structure. The diesel engine and engine-mounted components are constructed in accordance with IEEE Standard 387. The following instrumentation is

provided for the DGECWS: low-pressure jacket water; low-level jacket water standpipe; low-temperature jacket water in/out; high-temperature jacket water in; high-temperature jacket water out trip; and high-high temperature jacket water out trip. All alarms are separately annunciated on the local diesel engine control panel which also signals a general diesel alarm in the control room. The high-high temperature alarm will trip the diesel engine only if the diesel is in the test mode. The trip signal is locked out in the emergency mode and the engine continues to run. The high-high alarm in the emergency mode alerts the operator to prepare to switch over to the redundant diesel. Therefore, the DGECWS complies with the guidelines of RG 1.9 as it relates to system protective interlocks. On this basis, the staff concludes that the DGECWS complies with the requirements of GDC 17 regarding the capability of the system to meet independence and redundancy criteria and GDC 44 regarding the provisions for cooling water systems that can transfer heat from safety systems during normal and accident conditions, that have redundancy, and that can withstand a single failure.

Corrosion-inhibiting agents will be added to the cooling water by means of a chemical pot feeder unit, which can be connected to the DGECWS when needed. Additives will be compatible with cooling water system materials and will ensure proper system performance by maintaining water chemistry in accordance with the manufacturer's recommendations. System components and piping are tested to pressures designated by appropriate codes. Inspection and functional testing are performed prior to operation. The COL applicant will submit testing and calibration frequencies to the staff for review. The DG will be tested in accordance with the guidance of RG 1.108. On this basis, the staff concludes that the DGECWS complies with the requirements of GDC 45 and 46 with respect to inspection and testability of the system.

In RAI Q410.125.a, the staff asked ABB-CE to address the verification of DG performance for extended periods under no-load and light-load conditions without degradation of performance or reliability. ABB-CE stated that this concern was covered in the response to RAI Q430.27. ABB-CE responded to RAI 430.27, stating that after discussing this subject with several DG vendors, idle speed operation for periods of one hour does not appear to cause gum and varnish deposits of a nature severe enough to result in DG malfunction, provided that after periods of operations, the DG runs loaded for a length of time in order to clean and flush these deposits out of the exhaust system (one vendor suggests 30 minutes at 40-percent load, this following a four and half hour period of operation at idle speed).

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The staff reviewed this response and found that it is acceptable for the DG vendor to specify the operation of the DG during extended periods of no-load and light-load conditions. However, because COL applicants may purchase DGs from a variety of vendors with various manufacturer recommendations, final resolution of this issue will be the responsibility of the COL applicant. COL applicants will be expected to adhere to the guidance of the SRP and the SRP recommends adhering to the recommendations of the manufacturer. This was designated as DSER COL Action Item 9.5.5-2. This issue was resolved on the basis that Section 8.3.1.1.4.11 of the CESSAR-DC states that the DG will be loaded after no-load operation per manufacturers' recommendations. The staff finds this acceptable. Therefore, DSER COL Action Item 9.5.5-2 is acceptable.

On the basis of the above review, the staff concludes that the design of the DGECS complies with the requirements of GDC 17 and 44 with respect to redundancy, physical independence, and cooling capability, and GDC 45 and 46 with respect to inspection and testability of the system. As discussed in Section 9.5.4.1 of this report, the system also complies with the requirements of GDC 2, 4, and 5, as well as the recommendations of NUREG/CR-0660. Therefore, the staff concludes that the DGECS complies with the guidelines of SRP Section 9.5.5 and is acceptable.

9.5.6 Diesel Generator Engine Starting Air System

The staff reviewed the DG engine starting air system (DGESAS) in accordance with the guidance of SRP Section 9.5.6. DGESAS is designed to provide fast-start capability for the DG engine by using compressed air to rotate the engine until combustion begins and the engine accelerates under its own power. Staff acceptance is based on complying with the requirements of GDC 2 for protection against natural phenomena, GDC 4 for protection from environmental and dynamic effects of equipment failure, GDC 5 for sharing of SSCs between units, and GDC 17 for availability of electric power systems. Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report.

The staff based its review on the descriptive information and system diagrams in the CESSAR-DC (primarily Chapters 8 and 9) and the information supplied by ABB-CE in response to the staff's RAIs.

There are two standby EDGs in the CESSAR-DC design. Each DG engine has a physically separate and independent DGESAS division. Thus, a single failure in one of the two divisions will affect only the associated DG. Each

DGESAS division consists of two compressors, two aftercoolers, two filter/drying units, two air receiver tanks, and injection lines and valves. Either one of the DGESAS divisions (including two air receiver tanks) is capable of supplying sufficient air for five successful engine starts without the use of the compressors.

The starting air compressors are powered from a non-Class 1E motor control center. During a loss of offsite power, the starting air compressors are powered from the AAC power source. The water-cooled aftercoolers reject the heat of compression from the compressed air to the CCWS. To minimize the accumulation of moisture, the DGESAS has a multi-stage drying and filtering unit located in the line between the aftercooler and the receiver tank. The air first passes through a cyclone moisture separator and is filtered before entering one of two desiccant drying towers (alternating between active and regenerative cycles). The air is then filtered a second time before entering the receiver tank. The DGESAS is designed to supply air with a dewpoint at least 5.6 °C (10 °F) lower than the lowest expected ambient temperature.

To minimize fouling of the starting air valves and filters with contaminants, drip-traps on the moisture separators and the air dryer pre-filter collect any oil carryover. Drains are also provided on the aftercooler and air receiver tanks. Periodic blowdown of the drip-traps and drain valves will minimize the buildup of contaminants. Strainers upstream of the starting air solenoid valves keep rust out of the diesels. Two starting air receiver tanks with sufficient capacity to allow five successive DG engine starts, without the use of the compressor, are provided for each diesel.

The DGESAS is ANSI Class 4 from the starting air compressor through the desiccant drying towers, and ANSI Class 3 from the starting air receiver tank inlet check valve to the engine connections. The diesel engine and engine-mounted components are constructed in accordance with IEEE Standard 387.

Each starting air receiver has a set of pressure switches that control the operation of the air compressor. Pressure gauges on the tanks have a low-pressure alarm. A separate pressure switch on the engine control panel alarms if the air receiver tank pressure falls to a low setpoint. An automatic lockout on the engine's manifold prevents engine start attempts if the air pressure drops to a specified value and an alarm alerts the operator to take corrective action. The lockout ensures that there will be sufficient reserve for a manual restart. Periodic testing and maintenance of instruments is controlled by a preventive maintenance program. This was designated as DSER COL Action Item 9.5.6-1. This issue was closed based on the fact that

testing and maintenance of all systems are done by the COL applicant. The staff finds this acceptable. Therefore, DSER COL Action Item 9.5.6-1 is acceptable.

The starting air receiver tanks also supply air at reduced pressure to the diesel automatic safety shutdown system. The automatic safety shutdown system monitors the DG engine's parameters and trips the DG when the manufacturer's recommended setpoints have been exceeded. Group A trips are active only during periodic testing of the DG to prevent damage to the DG and are locked out during the emergency mode. Group B trips (engine overspeed, generator differential, low-low lube oil pressure, and generator voltage-controlled overcurrent) are active during all DG modes (testing and emergency). The low-low lube oil pressure trip is designed with a two out of three coincidence logic. The pneumatic logic for Group A and B trips consumes negligible volume because it operates on pressure rather than on flow capacity. There is sufficient air pressure available for operating the pneumatic logic after five successive start attempts. This complies with Position C.1.8 of RG 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used As Class 1E Onsite Electric Power Systems at Nuclear Power Plants." On this basis, the staff concludes that the DGELOS complies with the requirements of GDC 17 with respect to the capability of the system to comply with independence and redundancy criteria.

Several design parameters have been identified as future requirements to be determined once a DG vendor has been selected. Requirements to be specified include the devices to crank the engine, air start requirements with regard to the duration of the cranking cycle, and the number of engine revolutions per start attempt. These requirements will, in turn, dictate such design parameters as the volume and design pressures of the air receivers (sufficient for five diesel engine start cycles) and compressor size (capacity to recharge the system within 30 minutes). This information should be submitted with the COL application. This was designated as DSER COL Action Item 9.5.6-2. ABB-CE has included a sentence in Section 9.5.6.2.2 of Amendment Q to the CESSAR-DC which addressed the COL applicant providing specific DGELOS interface requirements. The staff finds this acceptable. Therefore, DSER COL Action Item 9.5.6-2 is acceptable.

On the basis of the above review, the staff concludes that the design of the DGELOS complies with the requirements of GDC 17 with respect to availability of electric power systems. As discussed in Section 9.5.4.1 of this report, the system also complies with the requirements of GDC 2, 4, and 5, as well as the recommendations of NUREG/CR-0660. Therefore, the staff concludes that the DGELOS

complies with the guidelines of SRP Section 9.5.6 and is acceptable.

9.5.7 Diesel Generator Engine Lube Oil System

The staff reviewed the DG engine lube oil system (DGELOS) in accordance with the guidance of SRP Section 9.5.7. The DGELOS is designed to deliver clean lubricating oil to the DG engine, its bearings and crankshaft, and other moving parts. By means of heaters, the DGELOS is designed to deliver warmed oil to the engine during standby to ensure fast-starting and load-accepting capability. The system also provides a means by which used oil may be drained from the engine and its components and replaced with clean oil. Staff acceptance is based on complying with the requirements of GDC 2 for protection against natural phenomena, GDC 4 for protection from environmental and dynamic effects of equipment failure, GDC 5 for sharing of SSCs between units, and GDC 17 for availability of electric power systems. Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report.

The staff based its review on the descriptive information and system diagrams in the CESSAR-DC (primarily Chapters 8 and 9) and the information supplied by ABB-CE in response to the staff's RAIs.

There are two standby EDGs in the CESSAR-DC design. Each DG engine has a physically separate and independent DGELOS division. Thus, a single failure in one of the two divisions will affect only the associated DG. Each DG unit utilizes a "dry sump" lube oil system, in which the supply of lubricating oil for the engine is stored in a separate lube oil sump tank, independent of and set at a lower elevation than the engine crankcase, allowing accumulated oil to drain, by gravity, to the sump tank. Clean lube oil is added to the sump tank from a clean lube oil storage tank. Used oil is pumped from the sump tank to a used lube oil storage tank.

The engine-driven lube oil pump picks up oil from the sump tank through a built-in suction pipe with a foot valve and delivers the oil in sequence from the pump discharge, first to the oil pressure regulating valves, which limit the maximum pressure on the pump discharge, and then in series through the lube oil cooler, the full-flow lube oil filter, and finally to the full-flow lube oil strainer. From the strainer, the oil enters the engine's internal circulation system. During engine standby, the motor-driven prelube oil pump operates continuously to ensure complete filling of the DGELOS. Oil that is circulated by the prelube oil pump passes over a set of thermostatically controlled electric immersion heating elements before leaving the

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sump tank to keep the engine warm. From the prelube oil pump, the oil passes in series through the prelube oil filter and the prelube oil strainer and enters the engine's internal circulation system. A separate drip lube system provides a continuous, metered flow of oil to the turbocharger bearings during engine standby to ensure adequate bearing lubrication for startup.

During DG operation, heat is removed from the DGELOS by the lube oil cooler and rejected to the DG engine cooling water system. A 10-percent design margin is built into the cooler, ensuring that the system is maintained in accordance with the manufacturer's recommendation. The motor-driven lube oil transfer pump, prelube oil pump, and the electric immersion heaters are powered from the essential auxiliary power supply. The clean and used lube oil pump motors are powered from the station's normal auxiliary power supply.

The DGELOS is an ANSI Class 3 piping system except for the clean and used lube oil transfer system which is an ANSI Class 4 piping system. The two systems are separated by ANSI Class 3 isolation valves. The diesel engine and engine-mounted components are constructed in accordance with IEEE 387. The exterior of carbon steel tanks and other underground carbon steel components are coated and protected from corrosion by an impressed current cathodic protection system in accordance with NACE Standard SRP-01-69. During the surveillance intervals for sampling the lube oil in the storage tanks, any accumulated water will be removed.

The DGELOS has the following instrumentation and alarms: lube oil sump tank local level indicator, and low-level-alarm annunciator; full-flow filter local pressure gauges, and high-differential-pressure alarm; full-flow strainer high-differential-pressure alarm. The DG engine has temperature and pressure monitoring systems with separate alarm and trip switches to alert the operator to abnormal operating conditions. During standby, low prelube oil pressure is alarmed to alert operating personnel to take corrective actions. If a shutdown setpoint/alarm is exceeded while the engine is operating during the test mode, a diesel trip will automatically shut down the engine. In the emergency mode (i.e., LOOP or LOCA) this trip is locked out and the engine continues to run. The alarm alerts the operator to prepare to switch over to the redundant diesel for power. Only a low-low engine lube oil pressure shutdown/alarm will trip the diesel engine regardless of the diesel operating mode. The periodic testing and maintenance of diesel engine lube oil instruments will be controlled by a preventive maintenance program. Therefore, the DGELOS complies with the guidelines of RG 1.9 as it relates to system protective interlocks. On this basis, the staff concludes that the

DGELOS complies with the requirements of GDC 17 regarding the capability of the system to comply with independence and redundancy criteria.

Specific design criteria, such as pump flows, operating pressure, temperature differentials, cooling system heat removal capability, and electric heater characteristics, are dependent on the selection of a DG vendor. In the DSER, the staff stated that this information must be submitted with the COL application. This was designated as DSER COL Action Item 9.5.7-1. ABB-CE addressed this issue in Section 9.5.7.2 of Amendment Q to the CESSAR-DC by stating the COL applicant will submit specific design criteria for the DGELOS. The staff finds this acceptable. Therefore, DSER COL Action Item 9.5.7-1 is acceptable.

On the basis of the above review, the staff concludes that the design of the DGELOS complies with the requirements of GDC 17 with respect to availability of electric power systems. As discussed in Section 9.5.4.1 of this report, the system also complies with the requirements of GDC 2, 4, and 5, as well as the recommendations of NUREG/CR-0660. Therefore, the staff concludes that the DGELOS complies with the guidelines of SRP Section 9.5.7 and is acceptable.

9.5.8 Diesel Generator Engine Air Intake and Exhaust System

The staff reviewed the DG engine air intake and exhaust system (DGEAIES) in accordance with the guidance of SRP Section 9.5.8. The DGEAIES is designed to supply clean air for combustion to the DG engine and to dispose of the engine exhaust to the atmosphere. Staff acceptance is based on complying with the requirements of GDC 2 for protection against natural phenomena, GDC 4 for protection from environmental and dynamic effects of equipment failure, GDC 5 for sharing of SSCs between units, and GDC 17 for availability of electric power systems. Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report.

The staff based its review on the descriptive information and system diagrams in the CESSAR-DC (primarily Chapters 8 and 9) and the information supplied by ABB-CE in response to the staff's RAIs.

There are two standby EDGs in the CESSAR-DC design. Each DG engine has a physically separate and independent DGEAIES division. Thus, a single failure in one of the two divisions will affect only the associated DG. Each DGEAIES division has a two-pipe combustion-air intake. The air is drawn in through in-line air filters before

entering the turbocharger. The turbocharger, driven by the hot engine exhaust gases, compresses the intake air and forces it through the engine aftercooler. The heat of compression is rejected to the DG engine cooling water system. Following combustion, the exhaust gases pass through a two-pipe exhaust system. The water-jacketed exhaust manifold discharges to the engine-mounted turbocharger. The exhaust piping then joins to pass through a single exhaust silencer and exits the DG building. There are no active components in the DGEAIES.

Intake air for each DG engine is part of the air drawn into the DG building by the diesel building ventilation system. The intake opening for the diesel building ventilation system is located at a minimum of 6.1 m (20 ft) above grade level to minimize the intake of dust in the DG building. Two filters, one on each of the two intake lines to each engine, prevent dust ingestion to each DG engine. The diesel exhaust gases are discharged to the atmosphere in a direction away from the outside air inlet with sufficient separation to minimize the effects of exhaust gas drift to the outside air inlet. Normal ventilation is filtered to maintain engine room cleanliness. All interior surfaces in the DG building are painted to minimize concrete dust. All components are located within a seismic Category I structure (DG building) and all essential components are fully protected from floods, tornado missiles, internal missiles, pipe breaks and whip, jet impingement, and interaction with non-seismic systems in the vicinity.

Each DG engine unit has sufficient instrumentation and alarms to monitor the combustion intake and exhaust system. A multi-point recorder on the local generator control panel records temperatures of the individual cylinder exhausts and the inlet and outlet turbochargers. A pyrometer on the local generator panel monitors the temperature on each cylinder and compares it to the average temperature of the other cylinders. The pyrometer annunciates a high/low temperature alarm on the local diesel engine control panel and signals a general diesel trouble alarm in the control room if a cylinder temperature exceeds the average temperature differential setpoint, with the pyrometer automatic sequencer stopping to display the out-of-tolerance cylinder. A high or low exhaust temperature does not trip the engine. A manual advance is also provided on the pyrometer to allow each cylinder to be checked and the inlet and outlet turbocharger exhaust temperatures to be checked. The turbocharger temperatures do not affect the cylinder temperature averaging circuit. Therefore, the DGEAIES complies with the guidelines of RG 1.9 as it relates to system protective interlocks. On this basis, the staff concludes that the DGEAIES complies with the requirements of GDC 17 regarding the capability of the system to comply with independence and redundancy criteria.

ABB-CE did not specify air flow capacity. As with the other DG auxiliary systems, this design characteristic is dependent on the selection of a DG vendor. In the DSER, the staff stated that this information must be submitted with the COL application. This was designated as DSER COL Action Item 9.5.8-1. ABB-CE has included a sentence in Section 9.5.8.2.1 of Amendment Q to the CESSAR-DC which addresses the COL applicant specifying the DG air flow capacity. The staff finds this acceptable. Therefore, DSER COL Action Item 9.5.8-1 is acceptable.

On the basis of the above review, the staff concludes that the design of the DGEAIES complies with the requirements of GDC 17 with respect to availability of electric power systems. As discussed in Section 9.5.4.1 of this report, the system also complies with the requirements of GDC 2, 4, and 5, as well as the recommendations of NUREG/CR-0660. Therefore, the staff concludes that the DGEAIES complies with the guidelines of SRP Section 9.5.8 and is acceptable.

9.5.9 Diesel Generator Building Sump Pump System

The staff evaluated the DG building sump pump system (DGBSPS) according to the guidance of SRP Section 9.3.3. The DGBSPS was reviewed to ensure its capability to remove leakage and equipment drainage from the DG building and to protect the DG units from internal flooding caused by the maximum credible pipe rupture in the DG building. The DGBSPS was evaluated to determine whether it meets the following criteria: GDC 2 relative to safety-related portions of the system being capable of withstanding the effects of earthquakes; GDC 4 relative to the system's capability to withstand the effects of flooding and the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and GDC 60 relative to providing a means to control suitably the release of radioactive materials in liquid effluent, including anticipated operational occurrences.

The staff based its review on the descriptive information and system diagrams supplied in the CESSAR-DC (primarily Chapters 8 and 9) and the information supplied by ABB-CE in response to the staff's RAI.

There are two standby EDGs in the CESSAR-DC design. Each DG building contains two sump pumps located in the pit below the lube oil sump tank. The pumps start automatically on high sump water level and transfer the water to the nuclear annex nonradioactive FDS. The DG building sumps and sump pumps are designed for a constant inflow rate of 284 L/min (75 gpm) with a maximum pump cycle time of three starts per hour. The maximum pumping flow rate with both pumps operating is 568 L/min (150 gpm).

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All components and piping of the DG building sump pump system are located within a seismic Category I structure (DG building) and all essential components are fully protected from floods, tornado missile damage, internal missiles, pipe breaks and whip, jet impingement and interaction with non-seismic systems in the vicinity. On this basis, the staff concludes that the DG building sump pump system complies with the requirements of GDC 4 with respect to withstanding the effects of flooding and the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

The DGBSPS is an ANSI Class 3 piping system and the pumps and system components are designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 3. Instrumentation associated with sump pump operation is Safety Class 3 and seismic Category I. The sump pumps are powered by the DG in the event of LOOP. Therefore, the design presented in the CESSAR-DC complies with Regulatory Positions C.1 and C.2 of RG 1.29 as related to the system being capable of withstanding the effects of earthquakes. On this basis, the staff concludes that the instrument air system complies with the requirements of GDC 2 with respect to seismic design.

The DGBSPS is connected to the nuclear annex nonradioactive EFDS upstream of that system's radiation monitors. The design prevents the inadvertent transfer of contaminated fluids to a non-contaminated drainage system for disposal and, therefore, complies with the requirements of GDC 60.

The adequacy of this design in regard to the maximum leakage or maximum credible pipe rupture in the DG building is a plant-specific issue and will be reviewed as part of a COL application. This was designated as DSER COL Action Item 9.5.9-1. ABB-CE has included a sentence in Section 9.5.9.2 of Amendment Q to the CESSAR-DC which states that the COL applicant will ensure the adequacy of the system with respect to the maximum leakage or maximum credible pipe rupture. The staff finds this acceptable. Therefore, DSER COL Action Item 9.5.9-1 is acceptable.

On the basis of the above review, the staff concludes that the design of the DG building sump pump system complies with the requirements of GDC 2, 4, and 60 with respect to seismic design, environmental conditions, and control release of radioactive materials, and is, therefore, acceptable.

9.5.10 Compressed Gas Systems

The compressed gas supply systems provide various gases for equipment and instrumentation cooling, purging, diluting, inerting, and welding. The gases are stored in high-pressure gas cylinders. Pressure regulators control the pressure of the gases as they are distributed throughout the plant. The various gases used are nitrogen, hydrogen, oxygen, carbon dioxide, argon, methane, and acetylene. All valves used for compressed gas service are designed and rated for gas service with resilient materials used for sealing surfaces.

The compressed gas supply systems serve no safety-related function. Failure of the system does not compromise any safety-related system nor does it prevent safe reactor shutdown.

In CESSAR-DC Section 9.5.10.1.2, ABB-CE states that the COL applicant shall ensure that the high pressure gas cylinders are not stored in areas that contain safety-related equipment so that the rupture of a cylinder does not adversely affect the operability of any safety-related component or equipment.

10 STEAM AND POWER CONVERSION SYSTEM

10.1 Introduction

The steam and power conversion system converts the heat energy generated by the nuclear reactor into electric power by using the heat energy to produce steam in two steam generators. The resultant steam is capable of driving a turbine generator unit. The steam and power conversion system for the System 80+ design is described in CESSAR-DC Chapter 10.

The steam and power conversion system uses a condensing cycle with regenerative feedwater heating. The turbine exhaust steam is condensed in a surface type condenser. The reference design includes a circulating water system using cooling towers. Alternative designs, as appropriate for each site, will be discussed in each site-specific application. The condensate from the steam is returned to the steam generators by means of the condensate and feedwater system (CFS).

A turbine bypass system (TBS) consists of eight turbine bypass valves and associated piping. The TBS relieves a combined capacity of 55 percent of total full power steam flow at 6895 kPa (1000 psia) to the condenser during startup, shutdown, load shedding, and transient conditions on the turbine, reactor, or both.

Each steam generator has two main steam supply lines to route the steam to the turbine generator unit. Each of the four lines has a flow-measuring device, five spring-loaded main steam code safety relief valves, a main steam isolation valve (MSIV), and a power-operated atmospheric dump valve (ADV). Each line combines to a common equalizer header. Downstream of the header, the four steam lines are routed to the turbine through a main steam stop valve, and a control valve (located just upstream of the high-pressure turbine). The main steam code safety relief valves provide overpressure protection for the shell side of the steam generators and the main steam piping up to the inlet of the main steam stop valves. The ADVs are used (1) as a heat sink for steam generator cooldown when the MSIVs are closed or the main condenser is not available, (2) to hold the plant at hot standby or to perform a plant cooldown during a loss-of-offsite power or station blackout (SBO) event, and (3) to provide for a controlled cooldown in the event of a main steam line break or steam generator tube rupture.

The steam and power conversion system consists of the following safety-related components:

- the emergency feedwater system (EFWS), including the main feedwater isolation valves (MFIVs) and associated piping to the steam generators

- the MSIVs, including the associated piping to the steam generators
- the ADVs
- the main steam code safety relief valves
- the steam supply to the EFWS

In the CESSAR-DC Section 10.1, ABB-CE describes the steam and power conversion system in detail. The system design and performance characteristics are identified in CESSAR-DC Table 10.1-1. The reference heat balance diagram, based on the use of a cooling tower, is identified in CESSAR-DC Figure 10.1-1. The main steam and feedwater system flow diagrams are shown in CESSAR-DC Figures 10.1-2, 10.3.2-1, and 10.4.7-1.

10.2 Turbine Generator System

The turbine generator system (TGS) converts the energy in steam from the nuclear steam supply into electrical energy. The TGS comprises all normally provided components and equipment, including the turbine main steam stop and control valves, and the intercept stop and control valves. The turbine generator consists of a double-flow, high-pressure turbine and three double-flow, low-pressure turbines driving a direct-coupled generator. The TGS has a turbine control and overspeed protection system to control all normal and abnormal operating conditions. This system includes redundant mechanical and electrical trip devices to ensure that a full-load rejection will not cause the turbine to overspeed beyond acceptable limits. The TGS is also designed so that the single failure of a main stop, main control, intercept stop, or intercept control valve does not disable the turbine overspeed trip function. CESSAR-DC Section 10.2 describes the TGS in detail. The design parameters are identified in CESSAR-DC Tables 10.2.2-1 and 10.2.4-1. CESSAR-DC Figure 10.1-1 depicts the reference flow diagram. TGS systems, structures, and components are classified in CESSAR-DC Table 3.2-1.

The electro-hydraulic control (EHC) system load control unit consists of two redundant turbine speed controllers (basic controllers). The turbine speed controllers, including the valve position controllers, use a "1-out-of-2" redundancy scheme. There is an automatic switchover between the controllers in case of a disturbance on one controller. One automatic controller provides a set value (such as load, frequency, pressure or limiters) to the basic controller, which provides positioning signals to the main control valves. The automatic controller also serves as the unit master interface. The load control unit provides (1) sensing functions to detect and generate signals proportional to unit loading, (2) limiting functions to constrain flow reference signals in response to signals

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from the sensing circuits, from the speed control unit, or from devices detecting the state of plant components, (3) Computing functions to separate flow reference signals for the valve sets, and (4) a logic function to ensure that proper permissives have been satisfied before the mode of operation is changed and to provide switching signals to EHC system devices.

There are four lines of defense against overspeed during all modes of operation: the two turbine speed controllers, a mechanical overspeed trip at 110 percent of rated speed, and electronic overspeed protection in a "2-out-of-3" logic scheme at 112 percent of rated speed. If the generator load is lost while the unit is in operation, an accelerator limiter (built into the EHC) senses the sudden load rejection and closes the control and intercept valves at the maximum rate. The entrained steam in the turbine casing, between the valves and the turbine, and in the crossover and extraction piping expands in less than 2 seconds. The expected overspeed is less than 10 percent (at full load), and the intercept valves reopen when the actual speed is below the set value. If the speed control on the control and intercept valves malfunctions when generator load is lost, the turbine accelerates until the mechanical overspeed trip activates to trip these valves and their actuators. The turbine then coasts down to zero speed. If the turbine continues to accelerate, the electronic overspeed protection, in "2-out-of-3" logic, actuates the tripping device. This causes the common safety system to be depressurized, and all stop and control valves close rapidly.

The turbine overspeed protection includes redundant mechanical and electronic overspeed protection. The mechanical overspeed trip device closes all stop and control valves by depressurizing both the hydraulic emergency system and, via an interface relay, the common hydraulic safety system. The mechanical overspeed trip device is set to activate at 110 percent of rated speed. The mechanical overspeed trip device is fed from the lube oil system; the interface between the lube oil system and the turbine safety system is made via a separating relay. The electronic overspeed protection uses the three binary signals from the speed conditioning units. However, these signals go to the "2-out-of-3" tripping device in the common safety system and not in the EHC. The electronic overspeed protection, which is set at 112 percent of rated speed, is provided as a backup to the mechanical overspeed trip device. Additionally, there are two redundant reverse power relays for tripping the generator breaker to prevent overspeed after turbine trip and to prevent overheating of the last two stages of low pressure (LP) turbine blades. Each of the reverse power relays has two strategies: (1) reverse power for more than 1 second and depressurize the turbine safety system; and (2) reverse power for more than 15 seconds. The turbine speed control system protection

devices are listed in CESSAR-DC Table 10.2.2-1.

The EHC system provides basic turbine control functions, including automatic control of turbine speed and acceleration through the entire speed range, and load and loading rate from auxiliary to full load. The EHC system provides fluid for turbine controls at 40 bar (580 psig). The turbine speed is measured by three independent speed modules including sensors and coordinating devices. The EHC is a microprocessor-based controller. The video-operated control room control panel keyboard or push button determines the increase or decrease inputs. The runbacks are determined by logic of speed control, load reference signal exceeding a preset load limit, loss of generator status coolant, process control system signal, and partial loss of load.

To prevent excessive decrease in steam throttle pressure, a main steam pressure limiter circuit closes the controlling valve set when the throttle pressure falls below a set point. The turbine and its control valves are designed to pass the rated steam flow at throttle pressures existing at the main steam stop valves at rated output of the nuclear steam supply system (NSSS). The load controller and the maximum load limiter are protected against overload. The stop valves are hydraulically operated from the common EHC system and all are equipped with limit switches for stroke testing. The control valve position loop consists of electronic circuitry, an electro-hydraulic servo-valve, a hydraulic actuator, and a linear position transducer. The position controller and the servo-valve are fail-safe, closing the control valve on loss of power. Each intercept valve is equipped with a position controller and a servo-valve. CESSAR-DC Section 10.2.2 describes the TGS and EHC system operation in detail.

The EHC equipment dc power supply (batteries) is redundant; independent ac power sources charge the batteries. The EHC system is divided into four safety categories to test the turbine safety system during operation:

(1) three channel sensing, normally energized and tested every four weeks during normal operation; (2) two channel sensing, normally deenergized and tested once per year during overhaul or during a standstill; (3) one channel sensing, normally energized and tested once a year during overhaul or during a standstill; and (4) one channel sensing, only for alarms.

The megawatt demand setter automatically coordinates turbine generator control with specific plant parameters to prevent exceeding NSSS limits affected by load transients, as described in Section 7.7 of this report.

The turbine safety system is independent of the turbine

control system. To protect the turbine generator, the following generated trip signals, in addition to the internal trips, will trip the EHC system:

- low condenser vacuum
- thrust bearing failure
- low bearing oil pressure
- generator internal fault
- generator breaker failure
- reactor trip
- loss of generator stator cooling without EHC runback
- steam generator high-high level
- safety injection
- all main feedwater pumps tripped
- manual turbine trip
- turbine oil fire trip
- moisture separator reheater drain tank high level
- excess vibration

The TGS uses main steam stop and control valves, and intercept stop and control valves. The turbine overspeed trips close these valves within pre-defined time periods after a trip signal to preclude an unsafe turbine overspeed condition. These closing times account for the residual steam in the piping between the valves and the turbine. In its response on January 24, 1992 to the staff's request for additional information (RAI) Q410.130, which is listed in Appendix B of this report, ABB-CE stated that the CESSAR-DC would be revised to include the closing time criteria for the extraction steam non-return check valves and the above turbine valves, in lieu of the actual closing time for these valves. In the staff's draft safety evaluation report (DSER), this was designated as DSER Confirmatory Item 10.2-1.

In CESSAR-DC, ABB-CE states that the closing times for full load rejection or turbine trip shall be such that the maximum expected overspeed of the turbine shall not exceed 110 percent of rated speed, and that overspeed control systems and turbine inertia shall be considered when establishing this criteria. In addition, ABB-CE states that the non-return check valves are capable of closing within a pre-defined time period to maintain stable turbine speeds in the event of a TGS trip. On this basis, DSER Confirmatory Item 10.2-1 is resolved.

Selection of the valve operation times that meet the above stated criteria was identified as a COL Action Item 10.2-1 in the DSER. In CESSAR-DC, ABB-CE states that the COL applicant will ensure that the selection of the turbine valve operation times meets the turbine valve closing/trip criteria. This remains COL Action Item 10.2-1.

The DSER identified as Confirmatory Item 10.2-2, that the CESSAR-DC would be revised to include the criteria that

the speed governor for normal speed load-control fully closes the control and intercept valve at 105 percent of normal turbine operating speed. In CESSAR-DC, ABB-CE states that the active speed governor fully closes all main and intercept control valves at 105 percent of normal turbine operating speed. The accelerator limiter is activated during a high load rejection, and these valves fully close below 105 percent. On this basis, DSER Confirmatory Item 10.2-2 is resolved.

Continuous access to the system components is possible during normal conditions. ABB-CE states that no radiation shielding is required for the components of the turbine generator and related steam handling equipment. The vacuum pump air discharge is continuously monitored for radiation to detect steam generator primary-to-secondary tube leaks, as discussed in Section 10.4.2 of this report.

The turbine generator is located in the turbine building and is separated from all portions of safety-related systems; no safety-related structures, systems, or components are located in the turbine building. Therefore, in the event of a failure of a high- or moderate-energy line, safety-related components will not be affected. The turbine orientation and the design of the safety-related structures provide protection against turbine missiles.

The program for in-service inspection of the main steam and intercept valves is described in CESSAR-DC Section 10.2.5. It includes, at intervals of approximately 3-1/3 years, during refueling or maintenance shutdowns coinciding with the inservice inspection schedule (required by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PV Code)) for reactor components: (1) dismantling of at least one main stop valve, one main control valve, one intercept stop and control valve; and (2) visual and surface examinations of the valve seats, disks, and stems. In addition, the following tests are performed during normal operation: (1) the main stop, control, and intercept stop and control valve are stroked from the control room during normal operation once every two weeks, and (2) a signal to allow the extraction line power assisted check valves to partially close will be simulated from the control room once per month.

The design of the TGS is acceptable and meets the requirements of General Design Criteria (GDC) 4, "Environmental and Dynamic Effects Design Bases," with respect to the protection of structures, systems, and components important to safety from the effects of turbine missiles. The system meets the requirements by providing a turbine overspeed protection system to control the turbine action under all operating conditions. This ensures that a full-load turbine reject will not cause the turbine to

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overspeed beyond acceptable limits and will not result in turbine missiles. Therefore, the staff concludes that the TGS conforms to GDC 4, staff positions, and industry standards, and can, therefore, perform its designed safety function. The TGS is therefore acceptable.

10.3 Main Steam Supply System

10.3.1 System Design

The main steam supply system (MSSS) transports steam from the secondary side of the NSSS to the turbine generator. Other safety-related or non-safety-related MSSS auxiliaries dissipate heat during cooldown, following a turbine and/or reactor trip and during main condenser unavailability; isolate steam generators, as necessary (as in containment isolation or post-loss-of-coolant accident (LOCA), for example); and provide overpressure protection for the shell side of steam generators and main steam piping. The safety-related portion of the system includes a portion between the steam generators down to and including the MSIVs. In accordance with CESSAR-DC Table 3.2-1, the essential portions of the MSSS are designed to Safety Class 2, seismic Category I, and are subject to the quality assurance requirements of 10 CFR Part 50, Appendix B. The system description, design parameters, and reference flow diagram are given in CESSAR-DC Section 10.3.2, Tables 10.3-2 and 10.3.2-1, and Figures 10.1-2 and 10.3.2-1, respectively. MSSS systems, structures, and components are classified in CESSAR-DC Table 3.2-1.

The MSSS conveys high-pressure steam generated in two steam generators through two lines for each steam generator to a high-pressure (HP) turbine. Extraction steam from the HP turbine heats the feedwater in the HP feedwater heaters. Each line is equipped with (1) five ASME Code spring-loaded main steam safety valves (MSSVs) to protect against overpressurization of the shell side of the steam generators and the main steam line piping; (2) an atmospheric steam dump valve, which is located between the safety valves and the MSIV; and (3) an MSIV and a bypass valve to establish positive isolation against both forward and reverse steam flow. All four of the steam-lines are routed to a main steam equalization header close to the HP turbine to balance steam pressure before entering the HP turbine. From the equalization header, branch lines to the steam reheaters reheat the steam from the HP turbine before entering the low-pressure (LP) turbine.

As stated in ABB-CE's response of January 24, 1992 to the staff's RAI Q410.131, the safety-related portion of the MSSS is contained in the main steam valve house

(MSVH), which is a seismic Category I structure. This structure is designed and located to protect against such environmental hazards as wind, tornados, hurricanes, floods, missiles, and the effects of high- and moderate-energy pipe ruptures. The flood protection, protection against internally generated missiles, and protection against externally generated missiles and high- and moderate-energy pipe breaks are evaluated in Sections 3.4.1, 3.5.1.1, 3.5.2, and 3.6.1 of this report. ABB-CE also stated that non-seismic portions of the system and other systems that may interact with the essential portions of the system are designed to seismic Category II requirements. ABB-CE needed to modify the text and figures of CESSAR-DC Chapter 10 to reference the MSVH and clearly identify the equipment located in these structures (including all safety-related valves in the main steam system). This was identified as DSER Open Item 10.3-1 in the DSER.

In CESSAR-DC Sections 10.3.3 and 10.4.7.3, ABB-CE states that the MSIVs, MSIV bypass valves, MSSVs, and ADVs, as well as MFIVs and feedwater regulating valves, are located within the seismic Category I designed MSVH. CESSAR-DC Table 3.2-1 shows that the above valves are located in the MSVH, and the MSVH areas are classified seismic Category I, Safety Class 3, and Quality Class 1. The MSVH areas are shown in CESSAR-DC Figures 1.2-8 and 3.8-5. On this basis, DSER Open Item 10.3-1 is resolved.

On the basis of this review, the staff concludes that the system meets the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," with respect to the ability of the safety-related portions of the system, and the structures housing them, to withstand the effects of natural phenomena. The system also meets the requirements of GDC 4 with respect to the safety-related portions of the system, and the structures housing them, to withstand effects of external missiles, and internally generated missiles, piping whip, and jet impingement forces associated with pipe breaks. This meets RG 1.29, "Seismic Design Classification," Positions C.1.a, C.1.e, C.2, and C.3, or C.1.f, C.2, and C.3; RG 1.115, "Protection Against Low Trajectory Turbine Missiles," Position C.1; and RG 1.117, "Tornado Design Classification," Appendix Positions 2 and 4.

The main steam piping and its supports and restraints withstand operating loads and design-basis events. The flow area of the main steam piping is sufficient to keep steam velocity below 46 m/second (150 ft/second). Main steam piping layouts that result in 90-degree elbows and miters are minimized. The system valves are designed so that no damage is incurred as a result of excessive force during closure under design conditions.

ABB-CE has addressed the potential for steam hammer and relief valve discharge loads. Techniques used to minimize such occurrences are as follows:

- The MSSV and ADV discharge piping is arranged and supported to minimize discharge loads such that the limiting loads are not exceeded for normal and relieving conditions.
- The main steam lines are headered together between the MSIVs and the turbine stop valves such that the pressure drop between each steam generator steam nozzle and the above equalization header are approximately equal. Additionally, the equalization header is sized to allow a full closure test, at 90 percent power, of one of the high-pressure turbine stop valves without imposing a severe pressure/load transient on one of the steam generators.
- The drainage system for main steam piping is designed to remove water before and during initial rolling of the turbine, as well as during shutdown.
- The MSSVs and ADVs are arranged such that any condensate in the line between these valves and the main steam lines drain back to the main steam lines.
- The MSSS is designed to accommodate steam hammer dynamic loads and relief valve discharge loads resulting from the rapid closure of system valves and safety/relief valve operation without compromising safety functions.

The valves are also periodically in-service tested for freedom of movement during plant operation in accordance with ASME Code Section XI, "Rules of Inservice Inspection of Nuclear Power Plant Components," Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants." However, in the DSER, the staff stated that ABB-CE had not mentioned any activity or program regarding personnel awareness in relation to potential occurrence of steam hammer dynamics. ABB-CE needed to clearly state that such a program was a required activity for the COL applicant and to provide guidance for developing plant operating and maintenance procedures that protect against a potential occurrence of steam hammer. This was identified in the DSER as DSER Open Item 10.3-2. In CESSAR-DC, ABB-CE states that the COL applicant shall provide plant operating and maintenance procedures that educate the plant personnel on the potential for the occurrence of the steam hammer and water entrainment, and the means to minimize such occurrences. On this basis, DSER Open Item 10.3-2 is resolved and the action to be taken by the COL applicant is COL Action Item 10.3-1.

Although the System 80+ design can be used at either single- or multiple-unit sites, ABB-CE states in CESSAR-DC Section 1.2.1.3 that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of the shared structures, systems, and components to perform their required safety functions. This was identified as COL Action Item 10.3-2 in the DSER. Upon further review, the staff determined that the design described in the CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the MSSS meets the requirements of GDC 5, and COL Action Item 10.3-2 is no longer necessary.

The MSSS is capable of providing heat sink capacity for the nuclear reactor, pressure relief for the shell side of the steam generator and the main steam lines up to the MSIVs, and steam to the steam-driven safety-related pumps necessary for safe shutdown. Therefore, the MSSS meets the requirements of GDC 34, "Residual Heat Removal," with respect to the system function of transferring residual and sensible heat from the reactor system. The modulating ADV (one for each main steam line) maintains the steam pressure below the lowest setting of the MSSVs during emergency shutdowns or hot standby conditions. Each of the two valves for a single steam generator is powered from a different Class 1E power source to meet the single-failure criterion for a main steam line break. The ADVs are powered from the Class 1E batteries, with each valve powered from a separate channel. The ADVs can be operated manually from the control room or remote shutdown panel following a safe shutdown earthquake coincident with the loss of offsite power. The ADVs can be manually closed and positioned. These valves have both analog position and open/close indication lights. This meets Branch Technical Position (BTP) RSB 5-1, "Design Requirements of Residual Heat Removal System," and the position in Issue 1 of NUREG-0138, "Staff Discussion of 15 Technical Issues Listed In Attachment To November 3, 1976, Memorandum From Director, NRR to NRR Staff, dated January 18, 1979."

Additionally, in response to RAI Q410.2 (June 30, 1988), ABB-CE stated that a safety-grade air supply will be provided to operate the ADV actuators, if the normal air supply becomes unavailable. The back-up might be a nitrogen air supply or one compliant with standard Type A American National Standards Institute/American Nuclear Society (ANSI/ANS) 59.3 (1984), "Safety Criteria for Control Air System." This information is not reflected in

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CESSAR-DC Section 10.3.2.3, and this was identified as a Confirmatory Item 10.3-1 in the DSER. ABB-CE states in CESSAR-DC Section 10.3.2.3.2.3 that (1) the ADVs are manually operable from the main control room or the remote shutdown panel, and (2) diverse sources of motive and control power (125 Vdc supply from the station batteries) to the ADVs are provided from separate sources to meet single failure criteria. The ADVs are designed with a return spring that causes the valve to fail closed on loss of motive power or loss of control signal. On this basis, Confirmatory Item 10.3-1 is resolved.

Following a steam line break, either all steam paths downstream of the MSIVs are isolated by their respective control systems following a main steam isolation signal (MSIS), or the results of a blowdown through a non-isolated path are shown to be acceptable. An acceptable maximum steam flow from a non-isolated steam path is 10 percent of maximum steam rate (3.6×10^6 kg/hour (19×10^6 lb/hour) at 6895 kPa (1000 psi) saturated steam conditions). As stated above, non-seismic portions of the system and other systems that may interact with the essential portions of the MSSS are designed to seismic Category II. Therefore, failure of non-seismic systems, structures, or components will not preclude operation of the safety-related portions of the MSSS.

Detection of steam leakage from the system in the event of a steam line break is provided by the MSIS, which is a part of the engineered safety feature actuation system (ESFAS). This system actuates on steam generator pressure and level and containment pressure (high) signals. The MSIS actuates the MSIVs, the MSIV bypass valves, and the isolation valves between the MSIVs and the steam generators to limit steam blowdown resulting from a steam line rupture or component malfunction. The system instrumentation and controls are evaluated in Chapter 7 of this report. The MSIV closure time is described in CESSAR-DC Sections 10.3.2.3.2 and 10.3.4(F). ABB-CE revised CESSAR-DC Section 5.4.5.3 to include up to a 5-second MSIV closure-time capability to isolate the steam generators upon receiving a signal from the ESFAS.

The system design conforms with the Commission regulations given in GDC 2, 4, 5, and 34, as discussed above. Therefore, the staff concludes that the MSSS design conforms with the acceptance criteria of SRP Section 10.3 and is, therefore, acceptable.

10.3.2 Secondary System Materials

The staff evaluated the steam and feedwater system materials in accordance with SRP Section 10.3.6. The areas of review included selection and fabrication of materials, fracture toughness of Class 2 and 3 components, and ABB-CE's approach to erosion/corrosion.

To address the open items in the DSER, ABB-CE will comply with RG 1.71, "Welder Qualification for Areas of Limited Accessibility," except for the welder performance qualifications. ABB-CE's proposed alternative is to assign only the most skilled welders to weld those joints with limited accessibility. These joints are also subject to the inspections required by the applicable code. These precautions should provide adequate assurance of the acceptability of joints welded under conditions of limited accessibility. ABB-CE has provided an acceptable alternative to the recommendations in RG 1.71. On this basis, DSER Open Item 10.3-3 is resolved.

In Amendment N, ABB-CE revised the CESSAR-DC to specify the main steam, hot reheat, condensate, main feedwater piping systems, and the heater drain piping systems upstream of the drain control valves are carbon steel or equivalent. However, extraction steam piping, heater drain piping downstream of the drain control valves, and other piping exposed to wet steam or flashing liquid flow will be chromium-molybdenum alloy steel, stainless steel, or equivalent. Stainless steel will be used as tubing in various heat exchangers and as cladding of tubesheets in low chloride environments, with alloy content increasing with increasing chloride levels. For chloride levels above 800 ppm, high concentrations of dissolved solids (above 1000 ppm), or water contaminated by sewage discharges, titanium tubing will be used. These material selections have been used in numerous plants for twenty or more years, and there have been no failures under the stated environments. Accordingly, ABB-CE has provided adequate assurance of the safety and structural integrity of these systems for the 60 year life of the plant. On this basis, DSER Open Item 10.3-4 is resolved.

ABB-CE identified in the CESSAR-DC the specific materials for the steam and feedwater system. Specifically, ABB-CE stated that carbon-manganese and chromium-molybdenum steels are to be used for the main steam and main feedwater systems. Specific grades are as follows:

Main Steam

ASME Class 2 Portion	SA 672 Gr. C70, C1. 22 SA 106 Gr. B or C	(> 24" NPS) (> 24" NPS)
B31.1 Portion	A 672 Gr. C70, C1. 22 A 106 Gr. B or C	(> 24" NPS) (≤ 24" NPS)

Main Feedwater¹

ASME Class 2 Portion	SA 106 Gr. B or C	(≤ 24" NPS)
B31.1 Portion	A 672 Gr. C70, C1. 22 or A 106 Gr. C A 106 Gr. B or C	(> 24" NPS) (≤ 24" NPS)

¹ In areas where erosion/corrosion degradation are expected to be excessive, low alloy steel piping (e.g. A335-P5, P11/12, & P22) may be substituted to minimize the effects and/or allow higher bulk fluid velocities.

In CESSAR-DC, ABB-CE addresses the issue of erosion degradation of piping systems. For carbon steel piping systems, the following methods to minimize erosion/corrosion are described in CESSAR-DC Section 3.6.3.1.2.1:

- (1) The bulk velocity is limited to prevent excessive erosion of the pipe wall. The following velocity guidelines are used for carbon steel piping:

Recommended Bulk Velocity Guidelines

<u>Service</u>	<u>Velocity</u>
Steam Piping	45.7 m/sec (150 ft/sec)
Water	4.6 m/sec (15 ft/sec)
Recirculation Lines (infrequent use)	6.1 – 7.6 m/sec (20 – 25 ft/sec)

- (2) Piping design and routing will be used to reduce susceptibility of the piping to pipe wall thinning.

ABB-CE proposed that engineering evaluations would be performed on a case-by-case basis using industry accepted methods, such as the Electric Power Research Institute's (EPRI's) CHECKMATE. It also noted that the feedwater and condensate system will be designed to avoid erosion damage.

ABB-CE's design and layout of piping systems considers

the effect on the piping material from fluid velocity, bend location, and the location of flash points. The staff believes ABB-CE has demonstrated an appreciation of water characteristics, piping configuration, and materials selection, and that these factors will mitigate erosion and corrosion of piping systems. On this basis, DSER Open Item 10.3-5 is resolved.

ABB-CE also provides in CESSAR-DC the corrosion allowance value for a 60-year plant design life and its technical basis for the determination of that value. Carbon steel is to be used for the main steam and main feedwater systems with a minimum corrosion allowance of 2.5 mm (.1 in.) per year. This corrosion allowance is greater than that used in the design of most current plants and should account for the 60-year design life of a System 80+ plant. For piping sections that are more susceptible to erosion/corrosion degradation, chromium-molybdenum or stainless steel are used per CESSAR-DC, Section 10.3.6.2.G.5. On this basis, DSER Open Item 10.3-6 is resolved.

ABB-CE specified that corrosion/erosion-resistant materials will be used in piping susceptible to corrosion or erosion. The primary method of corrosion/erosion mitigation is prevention in the design stage, including the following design features:

- fluid velocity limits (CESSAR-DC Sections 10.3.2.3.1.D and 10.4.7.2.5.K)
- chrome-molybdenum or stainless steel used in highly

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susceptible piping systems with flashing or two-phase flow (CESSAR-DC Section 10.3.6.2.G.5)

The inspection programs described in the CESSAR-DC, Section 10.3.4, will provide further assurance that susceptible piping will not be excessively degraded by corrosion or erosion before it is detected. ABB-CE's inspection programs rely on the guidance provided in EPRI NP-3944, "Erosion/Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines." These guidelines have been accepted by the staff. On this basis, DSER Open Item 10.3-7 is resolved.

The design of System 80+ components will also address the potential influence of environmental effects on the fatigue life of materials over the 60-year design life. Recent fatigue test data in the following references indicate that the effects of the environment could significantly reduce the fatigue resistance of materials:

- K. Iida, et al., "Survey of Fatigue Strength Data of Nuclear Structural Materials in Japan," Abstract of Design-Basis Accident Committee Report, 1988. (Enclosure in letter dated July 2, 1991 from John W. Craig (NRC) to Edward Griffing (NUMARC))
- M. Higuchi, and K. Iida, "Fatigue Strength Correction Factors of Carbon and Low-Alloy Steels in Oxygen-Containing High-Temperature Water," Nuclear Engineering and Design, Volume 129, 1991, pp. 293-306
- J.B. Terrell, "Effect of Cyclic Frequency on the Fatigue Life of ASME SA-106-C Piping Steel in PWR Environments," Journal of Materials Engineering, Volume 10, Number 3, 1988, pp. 193-203).

The specific concerns relate to the reactor water and temperature environment and its synergistic interactions with the strain rate.

The issue of environmental effects on fatigue is currently under consideration by a special steering committee for cyclic life and environmental effects in nuclear applications of the Pressure Vessel Research Council (PVRC). These activities were initiated based on requests from the ASME B&PV Code committee and the Board on Nuclear Codes and Standards (BNCS). The charter of the PVRC steering committee is to provide guidance and direction related to determining the effects of light-water reactor service environments on the cyclic life properties of applicable materials. The steering committee is also evaluating application methodologies that include these effects in the fatigue analysis process.

Preliminary recommendations were provided to the BNCS in September 1992. The initial findings reported to BNCS were that the current fatigue (S/N) curves should be appropriate for PWR environments. Since the BNCS is not in complete agreement with the steering committee's position, this issue has not been resolved. ABB-CE will continue to monitor the industry activities related to fatigue curves and future fatigue analysis methodology.

System 80+ components will be designed to ASME B&PV Code rules. If the influence of environmental effects has not been incorporated into the Code rules at the time of the design, the potential effects will be addressed based on the technical understanding of the materials data and anticipated operating conditions. In CESSAR-DC Section 10.3.6.2, ABB-CE commits to account for the effects of the environment in the fatigue analyses, based on ASME Code requirements. On this basis, DSER Open Item 10.3-8 is resolved.

During the development of the DSER, the staff raised the issue that carbon steel materials may be susceptible to dynamic strain aging (DSA), which reduces the material fracture properties (C.W. Marschall, M.P. Landow, and G.M. Wilkowski, "Effect of Dynamic Strain Aging on Fracture Resistance of Carbon Steels Operating at Light-Water-Reactor Temperatures," ASTM STP 1074, ASTM, Philadelphia, PA, 1990, pp. 339-360). The staff held several discussions with ABB-CE on this issue. In its submittal of January 20, 1993, ABB-CE noted that industry studies are in progress to investigate the susceptibility of materials to DSA and the extent to which the fracture toughness properties are affected. The materials used in nuclear power plant engineered safety systems (carbon and alloy steels, stainless steels, and nickel base alloys) have adequate fracture toughness, either as an inherent property of the material (austenitic stainless steels and nickel alloys) or by ASME Code required Charpy v-notch testing of carbon and low-alloy steels. These materials have been extensively used in existing nuclear power plants and have performed successfully without failure. The staff concludes that, by meeting the ASME Code requirements, these materials will have an acceptable level of fracture toughness to account for DSA. On this basis, DSER Open Item 10.3-9 is resolved.

No copper alloys are used for components that are in contact with feedwater, steam, or condensate. Oxygen-induced corrosion is minimized by using corrosion-resistant materials for the steam reheaters, feedwater heaters, and the condenser. Main steam piping, hot reheat piping, condensate piping, feedwater piping, and heater drain piping upstream of the drain control valves are carbon steel. Extraction steam piping, heater drain piping downstream of the drain control valves, and other piping

exposed to wet steam or flashing liquid flow are chromium-molybdenum or stainless steel.

The staff concludes that the main steam and feedwater system materials are acceptable and meet the relevant requirements of 10 CFR 50.55a, GDC 1, "Quality Standards and Records," GDC 35, "Emergency Core Cooling," and 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This conclusion is based on the following considerations:

1. ABB-CE has selected materials for Class 2 and 3 components of the steam and feedwater systems that satisfy Appendix I of Section III of the ASME B&PV, and meet the requirements of Parts A, B, and C of Section II of the Code. In Section 5.2.1.1 of the CESSAR-DC, ABB-CE also met the recommendations of RG 1.85, "Materials Code Case Acceptability ASME Section III, Division I," which describes acceptable code cases that may be used in conjunction with this industry standard.
2. The ASME Code imposes fracture toughness requirements for ferritic steel materials in Class 2 and 3 systems. The fracture toughness tests, chemical composition, and mechanical properties required by the Code provide reasonable assurance that ferritic materials will have adequate safety margins against the possibility of nonductile behavior or rapidly propagating failure.
3. ABB-CE meets the regulatory positions or provides and meets an alternative to the regulatory positions in RG 1.71 that the staff has reviewed and found acceptable. The onsite cleaning and cleanliness controls during fabrication satisfy the position given in RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."

10.4 Other Features

10.4.1 Main Condenser

The main condenser (MC) is a non-safety-related component located in the turbine building. The MC functions as a heat sink for the steam and power conversion system. The MC is designed to condense the low pressure turbine exhaust steam so it can be efficiently

pumped through the steam cycle. The MC also serves as a collection point for feedwater heater drains and vents, CFS makeup, and miscellaneous equipment drains and vents. The MC hotwells serve as a storage reservoir for the CFS with sufficient volume to supply maximum condensate flow for 5 minutes. The MC is also designed to remove noncondensable gases from the condensing steam by the main condenser evacuation system (MCES). Heat is removed from the MC by the condenser circulating water system (CCWS). The MC system includes all components and equipment from the turbine exhaust to the connections and interfaces with the main condensate and other systems.

In the DSER, the staff stated that ABB-CE had not submitted system drawing(s) and a component design parameters table in CESSAR-DC Section 10.4.1. This was identified as DSER Open Item 10.4.1-1. ABB-CE now provides a system description, representative design parameters, and a reference flow diagram in CESSAR-DC Section 10.4.1, Table 10.4.1-1, and Figure 10.4.1-1, respectively. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 by identifying the MC as non-nuclear safety (NNS) and non-seismic category, and by indicating that the quality assurance requirements of 10 CFR Part 50, Appendix B, are not applicable. On this basis, DSER Open Item 10.4.1-1 is resolved.

In the event of a load rejection, the MC condenses up to 55 percent of the full-load main steam flow bypassed directly to the condenser by the TBS without tripping the reactor (refer to Section 10.4.4 of this report). If the MC is unavailable during a normal plant shutdown, a sudden load rejection, or a turbine trip, the spring-loaded safety valves can discharge full main steam flow to the atmosphere to protect the MSSS from overpressure. The MC removes non-condensable gases from the condensing steam through the MCES (refer to Section 10.4.2 of this report). The design also deaerates any drains that enter the condenser. Condenser tube material shall be as follows:

- Type 304L stainless steel tubing or equivalent for chloride levels below 200 ppm
- 316L stainless steel tubing or equivalent for chloride levels up to 500 ppm
- 904L stainless steel, AL-6X, or equivalent for chloride levels between 500 and 800 ppm
- titanium tubing or equivalent for greater than 1000 ppm of dissolved solids or chloride levels above 800 ppm.

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The main condenser is initially tested in accordance with the Heat Exchanger Institute standards for steam surface condensers. The condenser shells, hot wells, and waterboxes have access openings. Periodic visual inspections and preventive maintenance are conducted following normal industrial practice.

Although the design has radioactivity monitors in the system to detect leakage into and out of the MC during normal operation, startup, and shutdown, the main condenser has no radioactive contaminants inventory. Radioactive contaminants can only be obtained through primary-to-secondary system leakage resulting from steam generator tube leaks. The steam generator blowdown system (SGBS) continuously samples the radioactivity of the steam generator blowdown (refer to Section 10.4.8 of this report and CESSAR-DC Table 9.3.2-1, which gives process sampling requirements for normal operation). Since the radioactivity is continuously monitored to detect leakage into and out of the condenser at the vacuum pumps discharge, GDC 60, "Control of Releases of Radioactive Materials to the Environment," is met with respect to failures in the system design that could result in excessive releases of radioactivity to the environment. The vacuum pump discharge is continuously monitored for radiation in order to detect steam generator primary-to-secondary tube leaks (refer to Section 10.4.2 of this report). The radiological monitoring capabilities are discussed in Section 11.5 of this report. The main condenser is non-safety-related, serves no safety function, and is not required for any safety shutdown. Flooding as a result of an MC failure (loss of water box or circulating water piping) is limited to the turbine building. High-level alarms in the turbine building sump alert operators in the event of leaks large enough to flood the building. The operator can then isolate leakage paths and limit flooding. Additionally, no safety-related structures, systems, or components are located in the turbine building. Refer to Section 3.4.1 of this report for a discussion of flooding.

The MC design conforms with the Commission regulations given in GDC 60 with regard to failures in the system design that do not result in excessive releases of radioactivity to the environment, and do not cause unacceptable condensate quality or flooding of areas housing safety-related equipment, as discussed above. Therefore, the staff concludes that the design of the main condenser complies with the acceptance criteria of SRP Section 10.4.1 and is, therefore, acceptable.

10.4.2 Main Condenser Evacuation System

The MCES is a non-safety-related system located in the turbine building. The MCES removes air and other noncondensable gases from the condenser and maintains

adequate condenser vacuum for proper turbine operation during startup and normal operation. The MCES includes equipment and instrumentation to establish and maintain condenser vacuum and to prevent an uncontrolled release of radioactive material to the environment. The major components of the MCES are the vacuum pumps that are used to create a vacuum in the MC. The system equipment is NNS, non-seismic category, and is not subject to the quality assurance requirements of 10 CFR Part 50, Appendix B.

In the DSER, the staff stated that ABB-CE had not provided sufficient details (such as a detailed system description, flow diagram, piping and instrument diagram, and tabulation of the design parameters for the system components) for the staff to complete its review. ABB-CE gave little system information for the MCES; missing information included such items as the number of vacuum pumps and the manner in which the system is operated. The lack of the above system information was designated as DSER Open Item 10.4.2-1.

ABB-CE provides a system description and a reference flow diagram in CESSAR-DC Section 10.4.2 and Figure 10.4.2-1, respectively, as well as a classification of MC systems, structures, and components in CESSAR-DC Table 3.2-1. The four packaged/skid-mounted vacuum pumps (one spare) are used for hogging and holding modes of operations. The hogging mode reduces the MC pressure to 13 to 25 mm (5 to 10 in.) of Hg absolute. The holding mode reduces MC pressure to its operating value. Each vacuum pump is sized for 34 m³/minute (1200 standard cubic feet per minute (scfm)) for hogging and 1.4 m³/minute (50 scfm) for holding modes in accordance with Heat Exchanger Institute standards for surface condensers. On this basis, DSER Open Item 10.4.2-1 is resolved.

The system design was originally based on the use of air ejectors. In CESSAR-DC Section 10.4.2, ABB-CE states that there is no direct connection between the main vacuum system and the reactor coolant system. Therefore, the normal function of one does not directly affect the other. The steam jet air ejectors (SJAEs) discharge is continuously monitored for radiation to detect steam generator primary-to-secondary tube leaks. However, CESSAR did not originally discuss the provisions for monitoring the discharge of the vacuum pumps, because these pumps might be used at different times than the SJAEs and may have a different discharge path. From the information in CESSAR-DC Section 10.4.2, the vacuum pump discharge path and/or the monitoring capabilities associated with this path were not identified. This was identified as Open Item 10.4.2-2 in the DSER. ABB-CE now provides a system description and a reference flow diagram in CESSAR-DC Section 10.4.2 and Figure

10.4.2-1, respectively. This information shows that vacuum pumps, rather than SJAEs, are used for system hogging and holding operations. The vacuum pump discharge is continuously monitored and routed to a unit vent in the nuclear annex. The radiological monitoring capabilities are discussed in Section 11.5 of this report. On this basis, DSER Open Item 10.4.2-2 is resolved.

The requirements of two Commission regulations (GDC 60, as it relates to the MCES design for the control of releases of radioactive materials to the environment, and GDC 64, "Monitoring Radioactivity Releases," as it relates to the MCES design for the monitoring of releases of radioactive materials to the environment) are considered met if the regulatory positions contained in the following RGs and Heat Exchange Institute standards are conformed to:

- (1) RG 1.26, "Initial Test Programs for Water-Cooled Nuclear Power Plants," as it relates to the MCES quality group classification that may contain radioactive materials, but is not part of the reactor coolant pressure boundary and is not important to safety
- (2) RGs 1.33, "Quality Assurance Program Requirements (Operation)," and 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," as they relate to the quality assurance programs for the MCES components that may contain radioactive materials
- (3) "Standards for Steam Surface Condensers, Heat Exchanger Institute," as it relates to the MCES components that may contain radioactive materials

In the DSER, the staff stated that ABB-CE had not demonstrated sufficient conformance to the regulatory positions for the staff to conclude that the system is acceptable. This was identified as DSER Open Item 10.4.2-3. ABB-CE states in CESSAR-DC Section 10.4.2.1 that the system conforms with the guidance of RGs 1.26 and 1.28, and the Hydraulic Institute standards for steam surface condensers to satisfy the requirements of GDC 60 and 64. As identified in CESSAR-DC Table 1.8-1, RG 1.33 is not applicable and RG 1.28 is applied to the MCES instead of RG 1.123. On this basis, DSER Open Item 10.4.2-3 is resolved.

The MCES complies with the requirements of GDC 60 and 64 with respect to the control and monitoring of releases of radioactive materials to the environment by providing a controlled and monitored MCES. The system meets the acceptance criteria of SRP Section 10.4.2 and is, therefore,

acceptable.

10.4.3 Turbine Gland Sealing System

The turbine gland sealing system (TGSS) includes the equipment and instruments that are a source of sealing steam to the annulus space where the turbine and large steam valve shafts penetrate their casings. The system prevents steam from leaking out and air from leaking in through the turbine shaft glands and through various steam valve stems. The TGSS provides a continuous supply of "clean" steam from the main or auxiliary steam systems to the main turbine shaft seal, main and intercept control valves, main and intercept stop valves, and turbine bypass valves. This sealing steam is provided by the auxiliary steam system during cold startup or emergencies. Once the steam generators are brought up to full pressure, the sealing steam source is switched from the auxiliary steam source to the main steam source and/or the high-pressure packings leakage source as the turbine is brought up to load during normal operation. Excess steam is discharged to the MC. The sealing steam keeps air from leaking into the steam cycle and potentially radioactive steam from leaking out of the steam cycle into the turbine building. The system returns the air-steam mixture to the turbine gland steam packing exhaustor and condenser. The gland steam condenser, maintained at a slight vacuum by the redundant blowers, condenses the steam, returns the drains to the MC, and exhausts the noncondensable gases to the atmosphere through the plant vent. The radiological monitoring capabilities are discussed in Section 11.5 of this report.

In the DSER, the staff stated that ABB-CE had not provided a system flow diagram, a piping and instrument diagram, and a table for the design parameters of the system components. This documentation was required for the staff to complete its review. The lack of system information was identified as Open Item 10.4.3-1 in the DSER. ABB-CE provides a system description and a reference flow diagram in CESSAR-DC Section 10.4.3 and Figure 10.4.3-1, respectively, and classifies the TGSS systems, structures, and components in CESSAR-DC Table 3.2-1. The design parameters of the system components will be evaluated on a plant-specific basis relevant to the system acceptance criteria for ABB-CE referencing the CESSAR-DC design. On this basis, DSER Open Item 10.4.3-1 is resolved.

The indication and alarm devices are provided locally and in the control room for steam seal header pressure, temperature, and flow as part of the normal instrumentation, but these devices are not required for safe shutdown conditions. The design must meet the requirements of GDC 60, as it relates to the TGSS design

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for the control of releases of radioactive materials to the environment, and GDC 64, as it relates to the TGSS design for the monitoring of releases of radioactive materials to the environment. The staff considers these requirements met if the regulatory positions contained in the following RGs and Heat Exchanger Institute standards are conformed to:

- (1) RG 1.26, as it relates to the quality group classification for the TGSS that may contain radioactive materials, but is not part of the reactor coolant pressure boundary and is not important to safety
- (2) RGs 1.33 and 1.123, as they relate to the quality assurance programs for the TGSS components that may contain radioactive materials
- (3) "Standards for Steam Surface Condensers, Heat Exchanger Institute," as it relates to components that may contain radioactive materials

In the DSER, the staff stated that ABB-CE had not demonstrated sufficient conformance to these regulations for the staff to conclude that the system is acceptable. This was identified as Open Item 10.4.3-2 in the DSER. ABB-CE states in CESSAR-DC Section 10.4.3.1 that the system conforms with the guidance of RGs 1.26 and 1.28 to satisfy the requirements of GDC 60 and 64. As identified in CESSAR-DC Table 1.8-1, RG 1.33 is not applicable and RG 1.28 is applied to TGSS instead of RG 1.123. On this basis, DSER Open Item 10.4.3-2 is resolved.

Based on the above information, the staff concludes that the turbine gland seal system meets the requirements of GDC 60 and 64 with respect to the control and monitoring of releases of radioactive materials to the environment by providing a controlled and monitored TGSS. The system meets the acceptance criteria of SRP Section 10.4.3 and is, therefore, acceptable.

10.4.4 Turbine Bypass System

The TBS bypasses up to 55 percent of the total full-power main steam flow at the normal full-power steam generator pressure of 6895 kPa (1000 psia). The TBS is a non-safety system located in the turbine building.

In conjunction with the reactor power cutback system, the bypass capacity is intended to allow for a full turbine load rejection of any magnitude without tripping the reactor or lifting primary or secondary safety valves. The TBS also controls NSSS thermal conditions when reactor power exceeds turbine power or to prevent the opening of safety

valves following a unit trip. The TBS can maintain the NSSS in zero power conditions. In response to the staff's RAI Q410.9 (dated June 30, 1988), ABB-CE stated that the system bypass capacity is consistent with reactor transient analysis (55-percent capacity input). The system provides pressure control during the loss of one of three feedwater pumps and transmits a control element assembly automatic motion inhibit signal when the turbine power and reactor power fall below selected thresholds. It also provides for manual control of reactor coolant system temperature during an NSSS heatup or cooldown. The system is intended to permit operation of the turbine bypass valves in a manner that minimizes valve wear and maintains valve control. The system provides for operation of the turbine bypass valves to limit the flow imbalance between condenser sections to the flow capacity of one valve when all turbine bypass valves and condenser shells are available. It also provides a redundant means of avoiding excessive steam release as the result of operator error or a single-component failure. The TBS provides a condenser interlock that blocks turbine bypass flow when unit condenser pressure exceeds the preset limit. The system is not required for the safe shutdown of the reactor and does not perform a safety function. The system description, system design and performance characteristics, and flow diagrams are provided in CESSAR-DC Section 10.4.4, Table 10.1-1, and Figures 10.1-2 and 10.3.2-1, respectively. The classification of TBS systems, structures, and components is provided in CESSAR-DC Table 3.2-1.

The TBS takes steam from the main steam header upstream of the turbine main stop valves and discharges it directly to the MC, bypassing the turbine generator. The system comprises all components and piping from the branch connection at the main steam system to the MC. The system consists of eight air-operated turbine bypass valves located in groups of four on the two steam lines, and associated piping and instrumentation. Normally, these valves are controlled by the steam bypass control system (SBCS) (refer to Section 7.7.1 of this report) but can also be controlled remotely or manually (local manual operation). The maximum capacity for a turbine bypass valve is approximately 10 percent of full steam flow at 6895 kPa (1000 psia). The turbine bypass valves fail closed to prevent uncontrolled bypass of steam to the condenser. The system valves modulate fully open or fully closed in a minimum of 15 seconds and a maximum of 20 seconds while in the automatic operation mode; they are designed to open in less than 1 second and close in 5 seconds in response to SBCS signals. The system can control steam flow as low as approximately 3.3 percent of total full-power steam flow during hot standby to permit pre-core hot function testing. The bypass control system is addressed in Section 7.7.1 of this SER.

Should the bypass valves fail to open, the secondary side ASME Code safety valves provide ultimate steam generator overpressure protection and main steam line overpressure protection. Should the condenser not be available, an interlock will prevent opening, or if opened, will close the system valves; operation of the system thus has no adverse effects on the reactor coolant system. The total system valve capacity (55 percent of total full-power steam flow at 6895 kPa (1000 psia)), in conjunction with the SBCS and reactor power runback system, allows a turbine full-load rejection without causing a reactor trip or lifting the primary or secondary code safety valves. Therefore, the staff concludes that the system is designed to enable sufficient steam to be bypassed to the MC so that the plant can be shut down during normal operation without using the turbine generator. The system therefore meets GDC 34, "Residual Heat Removal," with respect to the ability to use the system for shutting down the plant during normal operations.

ABB-CE considers the TBS a non-safety-related system and does not require any test or inspection of the system. However, according to SRP Section 10.4.4, preoperational and startup tests should conform with recommendations of RG 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors." The lack of such test requirements was identified as DSER Open Item 10.4.4-1. ABB-CE states in CESSAR-DC Section 10.4.4.4 that the system conforms with the recommendations of RG 1.26. For the preoperational and startup tests, a test will be conducted to verify opening of the turbine bypass valves in response to a signal, simulating turbine bypass, from the system bypass control system. On this basis, DSER Open Item 10.4.4.4-1 is resolved.

The system components are NNS, non-seismic category, and not subject to the quality assurance requirements of 10 CFR Part 50, Appendix B. The system is located in the turbine building, which is a non-seismic category building. No safety-related equipment is located within the turbine building or near the TBS. Therefore, the staff concludes that the system complies with the requirements of GDC 4 regarding adverse effects of a pipe break or malfunction on those components of the system necessary for shutdown or accident prevention or mitigation, as there are no such systems in the turbine building.

The as-built pressure drop between the steam generator nozzles and each system valve is site specific. The COL applicant referencing this CESSAR-DC will need to provide these pressure drops to evaluate the actual relieving capacity of those valves for the staff's review. In CESSAR-DC, ABB-CE states that the COL applicant shall provide the as-built pressure drop between the steam generator outlet nozzles and each turbine bypass valve to

enable ABB-CE to evaluate the actual relieving capacity. This is acceptable and is designated COL Action Item 10.4.4-1.

On this basis, the staff concludes that the system design meets the requirements of GDC 4 and 34, and that the TBS conforms with the applicable criteria of SRP Section 10.4.4 and can perform its intended function. The system design is, therefore, acceptable.

10.4.5 Condenser Circulating Water System

Although the majority of the CCWS is a non-safety-related system provided by the COL applicant referencing the System 80+ design, and is beyond the scope of this report, the parts of the system that are located in the turbine building are within the certified design scope. The referenced CCWS provides cooling water for the turbine condensers and rejects heat to the environment via the normal heat sink. (A cooling tower is one of the means of rejecting heat for the reference plant, although site-specific considerations may allow alternative means of heat rejection.) The CCWS comprises all components and equipment necessary for supplying the MC with a continuous supply of cooling water. The system description, representative design parameters, and reference flow diagram are provided in CESSAR-DC Section 10.4.5, Table 10.4.5-1, and Figure 10.4.5-1, respectively. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 by identifying the CCWS as NNS and non-seismic category; the quality assurance requirements of 10 CFR Part 50, Appendix B, are not applicable. The system is located partly in the yard and partly in the turbine building; these locations are non-seismic category areas. The CCWS is not required to perform any safety-related function.

ABB-CE provides interface requirements in CESSAR-DC Section 10.4.5.1.2 to ensure the adequacy of the System 80+ design. The system design minimizes the potential for water hammer by providing adequate filling and high-point venting. Valve opening and closing times are selected to minimize water hammer effects. The CCWS reference design, described below, includes four condenser circulating water pumps that are motor-driven, high-flow, low-head pumps. The pumps are located so that the available net positive suction head (NPSH) exceeds the required NPSH as defined in the IRs. Means are provided to prevent or detect and control flooding of safety-related areas so that intended safety functions of a system or component will not be precluded as a result of leakage from the CCWS. Flooding protection is provided to ensure that large leaks from CCWS piping do not result in the loss of all CCWS pumps. The system can maintain full load with one pump down, except potentially during

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the three hottest months of the year. The reference plant condenser includes a three-zone, multi-pressure condenser (three condenser shells). Valving is provided so that each of the two condenser waterbox flowpaths (with conductivity monitors in each condenser hotwell shell to detect leakage of circulating water) can be isolated and the unit can be operated at partial load, thus preventing the non-condenser-quality water from flowing into the CFS. The cooling tower has two 100-percent capacity makeup pumps which provide filtered makeup flow as needed. Instrumentation allows monitoring of system parameters, including condenser inlet and outlet temperature for each tube bundle, pump suction and discharge pressure, differential pressure across the pumps, and pH and conductivity for the circulating water. Flow meters are provided for each pump and for total system flow to the condenser.

In a letter dated January 24, 1992, ABB-CE responded to the staff's RAI Q410.133 (listed in Appendix B of this report) concerning the system capability to detect leaks and secure the system quickly and effectively. ABB-CE stated that a failure in the CCWS or MC large enough to cause flooding will be detected by high-level alarms in the turbine building sumps, and the operators will then isolate the appropriate equipment. Additionally, the pits below the main condensor hold more water than the hotwell condensate volume; flooding resulting from loss of condenser water box or circulating water piping is limited to the turbine building, which contains no safety-related equipment. ABB-CE committed to supply this information and this was designated as DSER Confirmatory Item 10.4.5-1. ABB-CE now states in CESSAR-DC Section 10.4.1.3 that (1) the ground elevation of the turbine building is above the finished plant grade level, (2) all turbine building interconnections and pipe tunnels are above the maximum internal flood level or sealed to prevent back-flooding, and (3) internal flooding in the turbine building is limited to non-safety-related equipment because the turbine building contains no safety-related equipment. On this basis, DSER Confirmatory Item 10.4.5-1 is resolved.

In CESSAR-DC Chapter 3, ABB-CE addresses generic flood protection criteria. However, ABB-CE did not specifically analyze the effects of a circulating water system failure, including an expansion joint failure. Flooding in the turbine building will not affect safety-related equipment, since none is located in this structure; however, it was not entirely clear that the flood protection measures given would keep the circulating water failure from flooding other structures to which the turbine building is connected.

Therefore, in the DSER, the staff could not conclude that ABB-CE had met the requirements of GDC 4 with respect

to the effects of discharging water that may result from a failure of a component or piping in the circulating water system. ABB-CE was requested to include the results of such an analysis in the description of the circulating water system in CESSAR-DC Section 10.4.5. This was designated as DSER Open Item 10.4.5-1. ABB-CE now provides IRs in CESSAR-DC Section 10.4.5.1.2.J, that means shall be provided to prevent, or detect and control, flooding of safety-related areas so that (1) the intended safety functions of a system or component will not be precluded due to leakage from the CCWS and (2) a malfunction or failure of a component or piping of the system shall not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components. The above requirements will be evaluated during the COL applicant review referencing the System 80+ design. Also, CESSAR-DC Section 3.4.4.1 states that the entrances to the nuclear annex from the turbine building are elevated above plant grade to prevent flood propagation. On this basis, DSER Open Item 10.4.5-1 is resolved.

In the DSER, the staff also stated that ABB-CE did not include the system flow diagram, the piping and instrument diagram, or a table giving the design parameters of the system equipment and associated sufficient system description for all modes of operation to allow the staff to complete its review. To complete its review, the staff also requested information addressing the normal operating conditions of the system (normal inlet and outlet temperature or flow capacity of the circulating water pumps). The lack of this information was identified as DSER Open Item 10.4.5-2. ABB-CE now provides representative design parameters and a reference flow diagram in CESSAR-DC Section 10.4.5 and Table 10.4.5-1, and Figure 10.4.5-1, respectively. The above plant-specific considerations will be evaluated during the COL applicant review referencing the System 80+ design. On this basis, DSER Open Item 10.4.5-2 is resolved.

Interface Requirements (IRs): In a letter of February 28, 1992, ABB-CE summarized the IRs for the System 80+ design that would be incorporated into CESSAR-DC Section 10.4.5. These IRs were identified as an Interface Item 10.4.5-1 in the DSER. ABB-CE now provides the IRs stated above in CESSAR-DC Section 10.4.5. The staff reviewed these IRs and finds them appropriate to be used by a reference applicant. Also, in CESSAR-DC Section 10.4.5, ABB-CE identifies a single cooling tower as one of the means of rejecting heat for the reference plant. However, site-specific considerations may allow for alternate means of transferring heat to the ultimate heat sink. The site-specific considerations also include additional details on the system (piping and instrumentation diagrams and component description tables) other than the

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safety portions of the systems are designed to Safety Class NNS, non-seismic requirements. As identified in CESSAR-DC Table 3.2-1, the valves, piping, and associated supports and restraints of the main feedwater system from and including the outboard MFIVs to the steam generator feedwater nozzles are seismic Category I and designed to Safety Class 2 (ASME Section III, Class 2) requirements; they are subject to the quality assurance requirements of 10 CFR Part 50, Appendix B. In the DSER, the staff stated that changes in safety classification were not indicated on the flow diagram (standard safety analysis report (SSAR) Figures 10.1-2 and 10.4.7-1). The piping and instrument diagrams should clearly indicate where class changes. This was identified as DSER Open Item 10.4.7-1 in the DSER. ABB-CE now provides representative flow diagrams in CESSAR-DC Figures 10.1-2 and 10.4.7-1 with the changes in safety classification. On this basis, DSER Open Item 10.4.7-1 is resolved.

The systems are designed such that adverse environmental conditions such as tornado missiles and floods will not impair their safety function. The safety-related components are designed to seismic Category I to withstand the effects of earthquakes and perform their safety functions to achieve and maintain safe shutdown conditions. The nonessential portions of the systems are designed in accordance with 1.29, "Seismic Design Classification" (Position C.2), and as stated in CESSAR-DC Table 3.2-1. Therefore, the staff concludes that the systems meet the requirements of GDC 2 with respect to safety-related portions being capable of withstanding the effects of earthquakes and other natural phenomena.

The systems meet the requirements of GDC 4 with respect to the environmental and dynamic effects associated with normal operating maintenance, testing, and postulated conditions. Components and piping are designed, protected from, or located to protect against the effects of high and moderate energy pipe ruptures, whip and jet impingement.

The feedwater system is protected against the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) by having the feedwater system designed in accordance with the guidance contained in BTP ASB 10-2, "Design Guidelines for Avoiding Water Hammers In Steam Generators," thereby reducing the possibility of water hammers in steam generators. In response dated January 24, 1992, to the staff's RAI Q210.92, ABB-CE supplied steam generator design information including the design features to help maintain the top feedwater (downcomer) line full of water. The System 80+ feedwater system and steam generators utilize both a downcomer and economizer feedwater line. Each

steam generator economizer and downcomer feedwater line nozzle is equipped with a 90-degree elbow, and check valves upstream of the feedwater line connections to the steam generators (refer to CESSAR-DC Figure 10.4.7-1 and Section 10.4.7.2.6). The requirement for tests to verify that unacceptable feedwater hammer will not occur using plant procedures is incorporated into CESSAR-DC Chapter 14. In a letter dated January 24, 1992, ABB-CE responded to the staff's RAI Q410.134 and committed to more fully address water hammer in CESSAR-DC Sections 10.4.7.2.5, 10.4.7.2.6, and 10.4.7.2.7, especially addressing the requirements for incorporating water hammer concerns into plant operating and maintenance procedures, system pipe routing, and selecting stroke times for the feedwater regulating valves. Incorporation of this information in CESSAR-DC was identified as Confirmatory Item 10.4.7-1 in the DSER. In CESSAR-DC, ABB-CE now states that (1) the COL applicant will make provisions to avoid water hammer including the development of system operating and maintenance procedures, (2) a 90-degree elbow facing downward is attached to each steam generator feedwater nozzle which would aid in the prevention of water hammer and deaerator, and connected piping is designed to prevent water hammer and (3) main feedwater pipe routing design and the feedwater regulating valve stroke times are such that water hammer is precluded. ABB-CE has adequately addressed feedwater control valve and controller designs with respect to water hammer potential and has added a COL action item to review operating and maintenance procedures to ensure that precautions taken will minimize or eliminate water hammers. On this basis, Confirmatory Item 10.4.7-1 is resolved.

Although the System 80+ design can be used at either single-unit or multiple-unit sites, ABB-CE states, in CESSAR-DC Section 1.2.1.3, that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5 with respect to the capability of shared structures, systems, and components to perform their required safety functions. This was identified as COL Action Item 10.4.7-1 in the DSER. Upon further review, the staff has determined that the design described in CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the CFS meets the requirements of GDC 5, and COL Action Item 10.4.7-1 is unnecessary.

The system has redundant MFIVs. Each valve actuator is physically and electrically independent. This meets the single-failure criterion for isolating feedwater piping

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safety portions of the systems are designed to Safety Class NNS, non-seismic requirements. As identified in CESSAR-DC Table 3.2-1, the valves, piping, and associated supports and restraints of the main feedwater system from and including the outboard MFIVs to the steam generator feedwater nozzles are seismic Category I and designed to Safety Class 2 (ASME Section III, Class 2) requirements; they are subject to the quality assurance requirements of 10 CFR Part 50, Appendix B. In the DSER, the staff stated that changes in safety classification were not indicated on the flow diagram (standard safety analysis report (SSAR) Figures 10.1-2 and 10.4.7-1). The piping and instrument diagrams should clearly indicate where class changes. This was identified as DSER Open Item 10.4.7-1 in the DSER. ABB-CE now provides representative flow diagrams in CESSAR-DC Figures 10.1-2 and 10.4.7-1 with the changes in safety classification. On this basis, DSER Open Item 10.4.7-1 is resolved.

The systems are designed such that adverse environmental conditions such as tornado missiles and floods will not impair their safety function. The safety-related components are designed to seismic Category I to withstand the effects of earthquakes and perform their safety functions to achieve and maintain safe shutdown conditions. The nonessential portions of the systems are designed in accordance with 1.29, "Seismic Design Classification" (Position C.2), and as stated in CESSAR-DC Table 3.2-1. Therefore, the staff concludes that the systems meet the requirements of GDC 2 with respect to safety-related portions being capable of withstanding the effects of earthquakes and other natural phenomena.

The systems meet the requirements of GDC 4 with respect to the environmental and dynamic effects associated with normal operating maintenance, testing, and postulated conditions. Components and piping are designed, protected from, or located to protect against the effects of high and moderate energy pipe ruptures, whip and jet impingement.

The feedwater system is protected against the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) by having the feedwater system designed in accordance with the guidance contained in BTP ASB 10-2, "Design Guidelines for Avoiding Water Hammers In Steam Generators," thereby reducing the possibility of water hammers in steam generators. In response dated January 24, 1992, to the staff's RAI Q210.92, ABB-CE supplied steam generator design information including the design features to help maintain the top feedwater (downcomer) line full of water. The System 80+ feedwater system and steam generators utilize both a downcomer and economizer feedwater line. Each

steam generator economizer and downcomer feedwater line nozzle is equipped with a 90-degree elbow, and check valves upstream of the feedwater line connections to the steam generators (refer to CESSAR-DC Figure 10.4.7-1 and Section 10.4.7.2.6). The requirement for tests to verify that unacceptable feedwater hammer will not occur using plant procedures is incorporated into CESSAR-DC Chapter 14. In a letter dated January 24, 1992, ABB-CE responded to the staff's RAI Q410.134 and committed to more fully address water hammer in CESSAR-DC Sections 10.4.7.2.5, 10.4.7.2.6, and 10.4.7.2.7, especially addressing the requirements for incorporating water hammer concerns into plant operating and maintenance procedures, system pipe routing, and selecting stroke times for the feedwater regulating valves. Incorporation of this information in CESSAR-DC was identified as Confirmatory Item 10.4.7-1 in the DSER. In CESSAR-DC, ABB-CE now states that (1) the COL applicant will make provisions to avoid water hammer including the development of system operating and maintenance procedures, (2) a 90-degree elbow facing downward is attached to each steam generator feedwater nozzle which would aid in the prevention of water hammer and deaerator, and connected piping is designed to prevent water hammer and (3) main feedwater pipe routing design and the feedwater regulating valve stroke times are such that water hammer is precluded. ABB-CE has adequately addressed feedwater control valve and controller designs with respect to water hammer potential and has added a COL action item to review operating and maintenance procedures to ensure that precautions taken will minimize or eliminate water hammers. On this basis, Confirmatory Item 10.4.7-1 is resolved.

Although the System 80+ design can be used at either single-unit or multiple-unit sites, ABB-CE states, in CESSAR-DC Section 1.2.1.3, that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5 with respect to the capability of shared structures, systems, and components to perform their required safety functions. This was identified as COL Action Item 10.4.7-1 in the DSER. Upon further review, the staff has determined that the design described in CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the CFS meets the requirements of GDC 5, and COL Action Item 10.4.7-1 is unnecessary.

The system has redundant MFIVs. Each valve actuator is physically and electrically independent. This meets the single-failure criterion for isolating feedwater piping

interconnecting the steam generators to preclude blowdown of both steam generators following a postulated pipe rupture. MFIVs are designed for complete termination of forward feedwater flow within 5 seconds of receiving an MSIS. Steam generator overfill due to a system malfunction is prevented by automatic closure of the MFIVs upon receiving a MSIS (signal is generated when the high steam generator water level setpoint is reached). The MFIVs in each feedwater line are remotely operated and capable of maintaining less than 1000 cc/hour leak rate. Also, redundant feedwater system check valves, in both economizer feedwater lines and the downcomer feedwater lines, provide abrupt, complete termination of an existing reverse flow condition (total reverse leak rate of the valves from each steam generator is less than 1000 cc/hour). Therefore, ABB-CE has met the requirements of GDC 44, "Cooling Water," by providing a redundant and isolable system capable of transferring heat loads from the reactor system to a heat sink under both normal and accident conditions. (The EFWS provides heat removal during accident conditions and is evaluated in Section 10.4.9 of this report.) On this basis, the CFS can provide sufficient cooling water to transfer the heat load of the reactor system under normal operating conditions and under accident conditions assuming loss of offsite power and a single failure. Portions of the system can be isolated so that the safety function of the system will not be compromised.

ASME Code Section III, "Rules for Construction of Nuclear Power Plant Components," and Section XI, "Rules for Inservice Inspections of Nuclear Power Plant Components," are used to inspect and test ASME Code Section III Class 2 piping. ASME Code Section III Class 2 valves are periodically inservice tested for exercising and leakage in accordance with ASME Code Section XI, Subsection IWV. Therefore, ABB-CE has met the requirements of GDC 45, "Inspection of Cooling Water System," by designing a feedwater system that permits inservice inspection of safety-related components.

In the DSER, the staff stated that ABB-CE had not included the functional testing of the systems and components to ensure structural integrity and leak-tightness, operability and performance of active components, and the capability of the integrated system to function as intended during normal, shutdown, and accident conditions. Therefore, compliance with GDC 46, "Testing of Cooling Water System," regarding functional testing of the CFS and its components, was identified as DSER Open Item 10.4.7-2. ABB-CE now states in CESSAR-DC Section 10.4.7.4 that the CFS testing includes the functional testing of the systems and components to ensure structural integrity and leak-tightness, operability and performance of active

components, and the capability of the integrated system to function as intended during normal, shutdown, and accident conditions. On this basis, DSER Open Item 10.4.7-2 is resolved.

In a letter dated January 24, 1992 (the response to the staff's RAI Q410.234 listed in Appendix B of this report), ABB-CE stated that the completion of detailed plant operating and maintenance procedures is outside the scope of design certification; however, a statement was added to the CESSAR-DC requiring that adequate provisions for avoidance of water hammer be provided in developing plant operating and maintenance procedures. The staff agrees with this approach since the referenced procedure is site-dependent. In CESSAR-DC, ABB-CE now states that the COL applicant shall develop the system operating and maintenance procedures for avoidance of water hammer for NRC review. This is COL Action Item 10.4.7-2.

The design of the CFS and supporting systems conforms with the Commission regulations given in GDC 2, 4, 5, 44, 45, and 46. The design of the CFS meets the guidance of SRP Section 10.4.7, and is, therefore, acceptable.

10.4.8 Steam Generator Blowdown System

The SGBS controls the quality of water on the shell side of the steam generators by removing chemical impurities and radioactive materials which accumulate as a result of primary to secondary and condenser tube leaks and corrosion of the steam generator materials. A continuous high-flow blowdown controls the concentration of these impurities.

Each steam generator has its own blowdown line with the capability of blowing down the hot leg and/or the economizer regions of the steam generator shell side. The system accommodates a continuous blowdown of either 0.2% or 1 percent of the maximum steam flow rate and up to 10 percent for a short period of time, not exceeding 2 minutes. The blowdown fluid passes to a flash tank where the flashed steam is separated and directed to the low-pressure feedwater heaters. The liquid portion is processed by passing it first through a heat exchanger and then through a blowdown filter where the major portion of solid impurities is filtered out. After filtration, the blowdown fluid is directed to cation and mixed-bed demineralizers where ionic species are removed. The processed fluid, which should meet the secondary water chemistry specifications, is then returned to the condenser. The system continuously processes the blowdown fluid at a rate of 0.2 percent of each steam generator's maximum steam flow, for full-power operation and normal steam generator chemistry, and 1 percent when the chemistry is outside the normal limits. There is a provision to isolate

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the portion of the blowdown system exiting the containment by the redundant blowdown line isolation valves that would close upon a MČIS containment isolation signal, or emergency feedwater actuation signal.

The staff reviewed the design for the SGBS in accordance with SRP Section 10.4.8. The scope of review included a piping and instrument diagram, seismic and quality group classifications, design process parameters, and ABB-CE's evaluation of the proposed system operation.

The SGBS design meets the primary boundary material integrity requirements of GDC 14 as it relates to maintaining acceptable water chemistry control during normal and anticipated operational occurrences by reducing corrosion of steam generator tubes and materials, thereby reducing the likelihood and magnitude of primary-to-secondary coolant leakage.

The SGBS is seismic Category I and ANS Safety Class 2 (which is equivalent to Quality Group B) from its connection to the steam generator inside the primary containment up to and including the first isolation valve outside the containment in accordance with RGs 1.26 and 1.29, since this portion of the SGBS is considered an extension of the primary containment. The SGBS downstream of the outer containment isolation valves is not safety-related and not seismic Category I, and meets the quality standards of Position C.1.1 of RG 1.143. The SGBS meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2.

ABB-CE revised the CESSAR-DC to specify that the portions of blowdown piping material with potential exposure to two-phase blowdown fluid will be made of stainless steel or other corrosion resistant material. On this basis, DSER Open Item 10.4.8-1 is resolved.

10.4.9 Emergency Feedwater System

The EFWS is an independent, dedicated, safety-related means of supplying secondary-side, quality feedwater to the steam generators for heat removal, and of preventing reactor core damage during emergencies. The system is designed to actuate automatically or manually in the event of a loss of normal feedwater, including a loss of normal onsite and offsite power. The system can initiate, following a major LOCA, with operator action to keep the steam generator tubes covered to minimize potential containment bypass leakage with a pre-existing condition of primary to secondary leakage. The system is located within the Nuclear Island (NI) structures, including the containment, reactor building, and nuclear annex. During normal plant operation, the system has no function; the

separate non-safety startup feedwater system with a non-safety-grade startup feed pump is used for normal startup and shutdown (refer to Section 10.4.7 of this report).

The EFWS includes all safety-related equipment from the emergency feedwater tanks to the connections with the steam generators. The EFWS contains two emergency feedwater storage tanks (EFWSTs). Each of the EFWSTs is intended to supply secondary coolant to one steam generator, although the capability to cross-connect the two EFWSTs by non-safety grade piping with a non-safety grade gravity feed condensate source for makeup is provided. Each EFWST is connected to one of the two steam generators through independent divisions of the EFWS consisting of two parallel pump lines. These pump lines discharge to a common header that injects into the main feedwater downcomer lines inside the containment as the last connection before the lines enter the steam generators. In each division of the EFWS, one pump is a full-capacity motor-driven pump and the second pump is a full-capacity turbine-driven pump. Steam for the turbine-driven pump is supplied by its respective steam generator. All controls, instrumentation, and valves that are essential to the emergency operation of the steam-driven pump subtrains are powered by battery-backed Class 1E power. The batteries are capable of powering the steam-driven pump subdivisions for an SBO of up to eight hours with appropriate load shedding. In addition to the batteries, an alternate ac source of standby power is provided for an extended SBO period. Each EFWS pump is capable of providing 100 percent of the required minimal flow of 1893 L/minute (500 gpm) plus an additional 644 L/minute (170 gpm) for recirculation flow, to meet the design-basis heat removal requirements. The divisional cavitating venturi in each common emergency feedwater (EFW) header restricts the output of two pump flow as well as the individual pump runout flow to the steam generator. The system description, system design and performance characteristics, and flow diagram are provided in CESSAR-DC Section 10.4.9, Tables 10.4.9-1 through 10.4.9-5, and Figure 10.4.9-1, respectively. The classification of systems, structures, and components is provided in CESSAR-DC Table 3.2-1 for the EFWS.

As identified in CESSAR-DC Table 3.2-1 (revised in response to the staff's RAI Q210.1), all components (including pumps, turbines, and tanks) and piping from, and including, the EFWSTs, up to the containment isolation valves are designed to be Safety Class 3 and seismic Category I. All components (including the cavitation venturi and its associated piping from, and including, the containment isolation valves to the steam generators, and the portion of the main feedwater lines used by the EFWS) are Safety Class 2 and seismic Category I. The EFWS components are located in seismic Category I

structures (the reactor building and the nuclear annex) which also protect the components from such external environmental hazards as tornadoes, hurricanes, floods, and external missiles. All essential components of the EFWS are (1) designed to account for, (2) located to protect against, or (3) otherwise protected from, the effects of internal flooding, internal missiles, effects of earthquakes, or the effects of high or moderate energy line breaks (as described in CESSAR-DC Chapter 3). The flood protection (high water levels), protection against the effects on internally and externally generated missiles, and protection against the effects of pipe whip and jet impingement due to high- and moderate-energy pipe breaks are evaluated in Sections 3.4.1, 3.5.1.1, 3.5.1.4, 3.5.2, and 3.6.1 of the DSER. The use of safety-grade dedicated EFWSTs eliminates any direct reliance on the non-safety-grade condensate storage system. However, connections to the condensate system do exist. These connections to the condensate storage system and the condensate cleanup and makeup sources (the demineralized water system) are all through Safety Class 3 isolation valves, with the exception of the return lines from the condensate cleanup to the EFWSTs and the overflow lines that are connected to the condensate storage system. These connections are located above the 1.3×10^6 L (350,000 gal) level in the EFWSTs. These provisions and design features comply with requirements of GDC 2 and 4 with respect to protection against the effects of earthquakes and pipe breaks and missiles. Acceptability of GDC 2 is based on meeting Position C.1 of RG 1.29 for safety-related portions and Position C.2 for non-safety-related portions.

Although the System 80+ design can be used at either single-unit or multiple-unit sites, in CESSAR-DC Section 1.2.1.3, ABB-CE states that the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants. In the DSER, the staff stated that should a multi-unit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5 with respect to the capability of shared structures, systems, and components to perform their required safety functions. This was identified as COL Action Item 10.4.9-1 in the DSER. Upon further review, the staff has determined that the design described in the CESSAR-DC does not share structures, systems, or components with other nuclear power units. Therefore, the EFWS meets the requirements of GDC 5, and COL Action Item 10.4.9-1 is unnecessary.

The EFWS can be actuated either manually or automatically by the ESFAS or an alternate feedwater actuation signal from the alternate protection system. The initiation signal starts all EFW pumps and supporting systems, aligns the EFW sources, and opens flow paths from the EFW pumps to the steam generators. The

instrumentation and controls are provided in the main control room and at the remote shutdown panel. This feature enhances the capability of system instrumentation and controls for prompt hot shutdown as is required by BTP RSB 5-1, "Design Requirements Of The Residual Heat Removal System." The instrumentation and controls provisions in the main control room and at the remote shutdown panel include (1) discharge and suction pressure, and discharge and recirculation flow for motor driven pumps; (2) suction pressure, turbine inlet pressure, turbine governor control speed, and discharge and recirculation flow for steam driven pumps; (3) temperature and level for EFWSTs; and (4) steam generator level. These and additional instrumentation and control provisions are identified in CESSAR-DC Table 10.4.9-4. ABB-CE does not discuss the availability of redundant instrumentation in order to limit or terminate EFW flow to a depressurized steam generator. However, each division of the EFWS contains a flow-limiting device and a cavitating venturi in the injection lines before the connection to the main feedwater downcomer lines. This venturi is sized to limit the flow to a damaged steam generator. The venturi begins to cavitate at 2461 L/minute (650 gpm) (with minimum EFWST level) and is designed to limit the EFW flow to 3028 L/minute (800 gpm) under maximum flow conditions (both pumps in the division operating, the EFWST at the maximum level, and the secondary side of the steam generator at atmospheric conditions). Since two separate EFWSTs are provided, one for each division, the flow to the intact steam generator in a main feedwater line break is unaffected by the amount of flow allowed to be transferred to the damaged steam generator. Therefore, the EFWS complies with the applicable requirements of GDC 19, "Control Room," criteria as related to the design capability of system instrumentation and controls for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown.

Redundancy of the system is maintained by using two independent divisions, each dedicated to one of the two steam generators. Within each division, two full-capacity pumps (one motor-driven and one turbine-driven) and two 100-percent-capacity EFWSTs are provided.

Diversity is provided by using two types of pump drives (steam turbines and electrical motors), and dc and emergency electrical power sources. The turbine-driven pump train is capable of operating independent from the availability of either onsite or offsite dc power. Steam supply valves for the turbine are air-operated and fail open on the loss of either air or dc power. The EFWS injection valves in the turbine-driven pump section of each division are dc powered, with each division receiving power from different dc power divisions. The motor-driven pumps are powered from Class 1E power supplies, and again, the

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pumps and associated valves in the two divisions are powered from separate diesel generator-backed divisions. CESSAR-DC Table 10.4.9-2 identifies the valve operators, type and sizes. The emergency power requirements for the controls, indications, and alarms for EFWS components are identified in CESSAR-DC Table 10.4.9-5.

Separation is maintained between divisions through the separation of the power supplies for each division and through the separation of the instrumentation and control subsystems by having appropriate measures which preclude interaction between subsystems. Also, the design incorporates independent piping subsystems, protected at interconnection points with appropriate isolation or check valves or both to ensure a high degree of piping separation, redundancy, and diversity. The two divisions are connected at only two points. The EFWSTs are connected through a non-safety-grade pipe that is isolated at each tank by a locked closed Safety Class 3 manual isolation valve. The discharges of the pump subtrains are cross-connected through a single Safety Class 3 pipe. This pipe is isolated from each pump subdivision by two Safety Class 3 locked closed manual isolation valves.

Each safety-grade EFWST contains 1.3×10^6 L (350,000 gal) sufficient inventory. This is to provide enough EFW flow to get the primary system to conditions that would permit the operation of the shutdown cooling system under the following conditions: feedwater flow to an unisolated steam generator with a feedwater line break for 30 minutes, refill of the intact steam generator, eight hours of operation at hot standby, and a six-hour cooldown to shutdown cooling conditions.

The design, therefore, enables the system to remove decay heat from the reactor coolant system under loss-of-offsite-power conditions, assuming a single active failure, and gives adequate isolation for the divisions of the EFWS (both from non-safety-related portions and from the other division). The design-basis heat removal requirements have also been adequately identified. Therefore, the system design meets the guidelines of BTP ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressured Water Reactor Plants," (regarding diversity of motive power), the guidelines of NUREG-0611, and the requirements of GDC 34 and 44 regarding the capability to transfer heat loads during normal operation and accident conditions assuming a loss-of-offsite power and any single active failure.

The descriptions of the protected locations of EFWS components, in CESSAR-DC Sections 10.4.9.1.2 and 10.4.9.2.2.3, did not address protection from sabotage. In CESSAR-DC Section 10.4.7.2.3, the ABB-CE states that the alternate ac source will provide backup power to the

startup feedwater pump. The staff considers that this additional redundancy reduces the risk of radiological sabotage associated with loss-of-offsite power compared to present reactors. Nevertheless, the staff considers (NRC Review Guideline 17, "Definition of Vital Areas," Rev. 1), that the seismic Category I EFWS is a vital system; therefore, as required by 10 CFR 73.55(c), access to all EFWS components essential to the emergency function, including power to motor-driven pumps, steamlines to steam-driven pumps, suction and discharge piping and valves, valve motor control centers, and EFW storage tanks, should require passage through two barriers. (Locked security doors controlling access between two adjacent vital areas are not desired, if access to each vital area is otherwise controlled). In the DSER, designation of the EFWS as a vital system was identified as DSER Open Item 10.4.9-1. ABB-CE now states in CESSAR-DC Section 10.4.9.1.2 that the EFWS system is a vital system. On this basis, DSER Open Item 10.4.9-1 is resolved.

Before initial plant startup, a comprehensive performance test is performed to verify that the design performance of the system and individual components are attained. After the plant is brought to operation, periodic tests and inspections of the EFWS components and subsystems are performed to ensure proper operation. These tests include a complete checkout at the component level during normal operation. Satisfactory operability of the complete system can be verified during the plant's refueling shutdown. To the extent practicable, the pumps, valves, heat exchangers, and piping in the system are designed and located to facilitate periodic inspection. All four pumps in the system have recirculation lines that will allow the operability testing of the pump during normal plant operation. The failure mode and effects analysis is presented in CESSAR-DC Table 10.4.9-3. These provisions comply with the requirements of GDC 45 and 46 regarding design provisions to permit periodic inservice inspection and appropriate functional testing of the system and components. However, the discussion in CESSAR-DC Chapter 10 did not discuss the 48-hour endurance test required to meet the guidance in NUREG-0611. Incorporation of this requirement into the inspection and tests section of CESSAR-DC Section 10.4.9 was identified as DSER Open Item 10.4.9-2 in the DSER. ABB-CE now states in Section 10.4.9.4.2 that the EFW pumps shall be tested, in accordance with the recommendations of NUREG-0611. A 48-hour endurance test will be performed to demonstrate that the pumps have the capability for continuous operation over an extended time period without failure. On this basis, DSER Open Item 10.4.9-2 is resolved.

As part of the probabilistic risk assessment (PRA) submitted for the System 80+, "PRA for the System 80+

Standard Design" (January 1991), ABB-CE performed a reliability analysis of the EFWS in accordance with the requirements of Item II.E.1.1 of NUREG-0737, "TMI Action Plan Requirements." ABB-CE evaluated the EFWS reliability for seven postulated transient and accident scenarios (a loss of main feedwater, a loss-of-offsite power, an SBO, the loss of a 416-V ac bus, the loss of a 125-V dc bus, the loss of one component cooling water train, and a main steam line break/steam generator tube rupture). Overall, system unavailability for each of these cases was determined using the methodology described in NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide," and NUREG/CR-2300, "PRA Procedures Guide," and using component failure rates from several generally accepted generic data bases. The staff primarily reviewed ABB-CE's reliability evaluation on the loss of main feedwater, loss-of-offsite power, and SBO scenarios.

The EFWS unavailability, as estimated in ABB-CE's PRA, meets the SRP acceptance criteria of 10^{-4} to 10^{-5} per demand. Only in the SBO scenario is the EFWS unavailability greater than 10^{-4} , at 3×10^{-3} . Such a relatively small number for this scenario is due to the existence of two turbine-driven pumps in the System 80+ design. It should be noted that ABB-CE did not take credit for the startup feedwater pump. The only identified weakness of the analysis is that common-cause failures of the injection valves were not handled in a comprehensive and consistent manner for all of the cases analyzed. However, the incorporation of a more comprehensive and consistent valve common-cause model would not affect the conclusions of the analysis.

The review of the EFWS reliability was based on the information contained in the PRA submitted as part of the CESSAR-DC. However, CESSAR-DC Section 10.4.9 references the reliability analysis presented in CESSAR-DC Appendix 10A. This appendix did not contain the same analysis as that submitted in the PRA, and did contain some modeling mistakes and different results from the PRA analysis, which is more recent. Incorporation of the more recent analysis into Appendix A, providing consistency between the Chapter 10 EFWS reliability analysis and the PRA analysis, was identified as Confirmatory Item 10.4.9-1 in the DSER. ABB-CE now provides the results from the PRA analysis (ABB-CE letter dated October 16, 1992) in CESSAR-DC Appendix 10 A and Section 19.6.3.7. On this basis, Confirmatory Item 10.4.9-1 is resolved.

In a letter of January 24, 1992, ABB-CE responded to the staff's RAI Q210.8 concerning the potential destructive water hammer during startup by giving the overview of the design criteria to minimize water hammer in the EFW

piping. These criteria include a 90-degree elbow at each steam generator nozzle; continuously rising EFW piping to each steam generator with the provision of a check valve in each EFW line; adequate filling and venting provisions; valve operating times in equipment specification; and preoperational testing of feedwater and EFW systems to preclude water hammer during startup, normal, and transient operations. ABB-CE stated that the detailed design (piping and support layout drawings, stress reports, operating procedures, etc.) is dependent on the vendor-supplied information and site specific details of the piping and plant design. IN CESSAR-DC, ABB-CE now states that the COL applicant will make information available on the provisions for avoidance of water hammer in the EFW piping for NRC review. This is COL Action Item 10.4.9-2.

ABB-CE committed to prepare a distribution system design guide to ensure that the final design for the COL applicant would be completed consistent with the design basis and methodology in the CESSAR-DC.

The staff agrees with this approach, since the additional information is site dependent and is to be provided by the COL applicant. Therefore, the staff will evaluate the EFW system piping issue on water hammer on a site-specific basis during COL review. ABB-CE's commitment to prepare the distribution system design guide concerning water hammer in EFW system piping was identified as Confirmatory Item 10.4.9-2 in the DSER. ABB-CE submitted the distribution system design guide for staff review and the guide was evaluated in Chapter 14 of this report. On this basis, Confirmatory Item 10.4.9-2 is resolved.

In a letter of January 24, 1992, responding to the staff's RAI Q730.4 concerning the steam binding of the EFWS pumps, ABB-CE stated that the temperature sensor located between the flow control valve and the isolation valve on each subtrain is continuously monitored and audibly alarmed in the control room. The COL applicant will include the provision of local monitoring of the temperature sensor upon loss of control room indication in its plant operating procedures. The temperature readings shall be recorded at least once per shift and before and after each EFW pump run. The system is designed for continuous system venting through the EFW storage tanks and by use of normally closed isolation valves upstream of the interface with the main feedwater system. The COL applicant's plant operating procedures will include the provision of the plant operator's action to vent the EFW pumps upon audible indication, as discussed above. These site-specific provisions provide guidance for recognizing the effects of steam binding of the EFW pumps and for restoring the system to operable status following a steam

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binding incident.

The staff agrees with the ABB-CE's guidance concerning the provisions for steam binding of EFW pumps and considers the surveillance and operating procedures site-specific. In CESSAR-DC, ABB-CE states that the COL applicant will make information available for NRC review on the provisions for avoidance of steam binding in the EFW pumps. This is acceptable and is designated COL Action Item 10.4.9-3.

The design of the EFWS and supporting systems meets the Commission's regulations given in GDC 2, 4, 5, 19, 34, 44, 45, and 46 with respect to protection against natural phenomena and environmental effects, shared systems, operational capability from the control room, decay heat removal and cooling water capability, and inservice inspection and functional testing; the guidelines of RG 1.29 and BTPs ASB 10-1 and RSB 5-1 concerning seismic classification, power diversity, and design of decay heat removal systems; and the recommendations of NUREG-0611 concerning generic improvements to the EFWS system design and reliability. The system conforms to the acceptance criteria of SRP Section 10.4.9 and is, therefore, acceptable.

10.4.10 Turbine Disc Integrity

The staff has reviewed the information submitted by ABB-CE in CESSAR-DC Sections 10.2.3 and 3.5.1 to ensure turbine disc integrity and a low probability of turbine disc failure with the generation of missiles. Standard Review Plan (SRP) Section 10.2.3, "Turbine Disk Integrity," provided guidance for review of this section. The areas reviewed included materials selection, fracture toughness, preservice inspection, turbine disc design, and in-service inspection. See Section 3.5 of this report. The staff's DSER Open Items 10.2-1 to 10.2-6 in the DSER are resolved as follows:

In CESSAR-DC, ABB-CE commits to perform Charpy testing in accordance with American Society for Testing and Materials (ASTM) A-370, "Standard Methods and Definitions for Mechanical Testing of Steel Products," and no exceptions were noted. This will provide adequate assurance that Charpy testing is performed correctly. On this basis, DSER Open Item 10.2-1 is resolved.

ABB-CE specifies the acceptance criteria for material fracture toughness for the low-pressure turbine disc material as a 50 percent fracture appearance transition temperature (FATT) at temperatures no higher than -18 °C (0 °F) for low-pressure turbine discs. The Charpy V-notch energy at the minimum operating temperature of each low-pressure disc in the tangential direction was

specified to be at least 82 J (60 ft-lb). On this basis, DSER Open Item 10.2-2 is resolved.

The fracture toughness, J_{Ic} , is determined by testing a three-point bend specimen per ASTM E-813. The material fracture toughness (K_{Ic}) value will be obtained from a relation between K_{Ic} and J_{Ic} in paragraph 9.4 of ASTM E-813. This test method for determining J_{Ic} is acceptable to the staff. ABB-CE has proposed that these tests be performed on the actual disc material at ambient temperature, with the added requirement that, for the specified material, the ratio of K_{Ic} to the maximum tangential stress at speeds from normal to design overspeed has to be at least 2.1 $\sqrt{\text{in}}$ for discs with the maximum operating temperature below 100 °C and 2.2 $\sqrt{\text{in}}$ for discs with the maximum operating temperature at or above 100 °C. SRP Section 10.2.3 states that this ratio be at least 2.0 $\sqrt{\text{in}}$ at operating temperature. The turbine discs should possess adequate fracture toughness, considering that (1) the fracture toughness of this material increases slightly with an increase in temperature, (2) the material might not be completely on upper-shelf at ambient temperature, and (3) the higher ratio requirement and the increase in ratio above 100 °C is an acceptable alternative approach. On this basis, DSER Open Item 10.2-3 is resolved.

In CESSAR-DC, ABB-CE describes the method of determining the yield strength of the material of the wheels and rotors. ABB-CE specified this to be the "minimum yield strength in the tangential direction." This ensures that adequate strength of these materials is provided in the weak direction of the materials. On this basis, DSER Open Item 10.2-4 is resolved.

Each finish-machined forging is subjected to 100-percent volumetric (ultra-sonic), surface, and visual examinations using the manufacturer's acceptance criteria. ABB-CE commits in CESSAR-DC to perform surface crack testing on all parts critical to rotor integrity, such as weld inspection holes, balancing holes, and geometric transitions. The disc is not keyed to the rotor in this design and high stress regions are limited in location; these commitments should ensure that in-service inspection of all high-stress regions is facilitated. On this basis, DSER Open Items 10.2-5 and 10.2-6 are resolved.

In describing the steam turbine design, ABB-CE states that unlike most domestic nuclear steam turbine rotor designs, the turbine does not use a shrunk-on disc design nor a one-piece forged rotor design. Instead, this turbine design uses barrel construction, which consists of a series of welded ring forgings. This design is in effect a one-piece, welded, hollow rotor design, with the end forgings having shaft protrusions. Each forging is chemically analyzed and mechanically tested for strength and toughness, and each

has an upset ratio of at least two. Each is separately heat treated and inspected before the final welding process.

Forgings to be used in the welded rotor are rough machined with minimum stock allowance before heat treatment. All individual rough machined rotor forgings to be used for the welded rotor are subjected to 100-percent volumetric (ultrasonic) examination. Because each forging is oversized, indications found during the ultrasonic testing may be removed during final machining of the forging. The manufacturer's acceptance criteria includes the requirement that subsurface sonic indications be evaluated to ensure that they will not grow to a size that compromises the integrity of the unit during the service life. Before welding, all surfaces that would not be accessible after welding will be surface examined. After welding and final machining, all surfaces exposed to steam will be surface examined, with particular attention placed on the areas of stress risers and welds. The welds will be ultrasonically examined in the radial and radial-tangential sound beam directions. On this basis, DSER Open Item 10.2-7 is resolved.

The turbine assembly is designed to withstand normal conditions and anticipated transients, including those resulting in a turbine trip, without loss of structural integrity. The highest anticipated turbine speed resulting from a loss of load is 110 percent of rated speed. The design overspeed is approximately 5 percent above this highest value (115 percent of rated speed). The turbine shaft bearings are designed to retain their structural integrity under normal operating loads and anticipated transients, including those leading to turbine trips. The natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20 percent overspeed are controlled in the design and operation to prevent distress to the unit during operation.

The maximum tangential stress in wheels and rotors resulting from centrifugal forces, interference fit, and thermal gradients does not exceed 0.75 of the yield strength of the materials at 115 percent of rated speed.

Each fully bucketed turbine rotor assembly is spin tested for three minutes at 120 percent of rated speed. This is above the maximum speed anticipated following a turbine trip from full load (110 percent).

The ratio of K_{Ic} at ambient temperature, to the maximum tangential stress for wheels and rotors at speeds from normal to 115 percent of rated speed, is at least 2.1 $\sqrt{\text{in}}$ for discs operating below 100 °C and at least 2.2 $\sqrt{\text{in}}$ for discs operating at 100 °C or above. Fracture toughness properties are obtained by destructively testing material taken from the rotor forgings (see the previous discussion

in the resolution of DSER Open Item 10.2-3).

Turbine wheels and rotors are made from vacuum-degassed Ni-Cr-Mo-V alloy steel by processes that minimize the occurrence of flaws and maximize fracture toughness. Phosphorous, sulfur, vanadium, and tin are controlled to reasonably low levels. Charpy tests are performed to determine the 50 percent FATT.

ABB-CE states in CESSAR-DC Section 3.5.1.3 that the COL applicant must submit to the NRC a summary of the turbine maintenance and inspection program and the results of the probabilistic calculations of turbine missile generation. In CESSAR-DC Section 10.2.4, ABB-CE describes the in-service inspection program for the turbine and the turbine preservice inspection and maintenance program. Turbine missile generation and probabilistic determination are discussed in Section 3.5.1.3 of this report.

ABB-CE's in-service inspection program meets SRP Section 10.2.3 and is acceptable. The in-service inspection program for the turbine assembly includes disassembly of the turbine's last two stages of blades in stages over an interval of approximately ten years or less during plant shutdowns, so that the entire turbine is inspected within ten years. This includes complete inspection of all normally inaccessible parts, such as couplings, coupling bolts, turbine shafts, low-pressure turbine buckets, low-pressure wheels, and high-pressure rotors. This inspection consists of visual, surface, and volumetric examinations.

ABB-CE has met the requirements of GDC 4 (10 CFR Part 50) with respect to the use of materials with acceptable fracture toughness and elevated temperature properties, adequate design, and the requirements for preservice and in-service inspections. ABB-CE has described its program for ensuring the integrity of low-pressure turbine discs by the use of suitable materials of adequate fracture toughness, conservative design practices, and preservice and in-service inspections. These provisions provide reasonable assurance that the probability of failure with missile generation is low during normal operation and transients up to design overspeed.

11 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description/Source Terms

The radioactive waste (radwaste) management systems for the ABB-CE System 80+ standard design control the handling and treatment of liquid, gaseous, and solid radioactive wastes.

Liquid radwaste from equipment and floor drains, reactor-grade laboratory drains, steam generator (SG) blowdown, condensate polisher regeneration, chemical wastes, laundry, and showers is collected and processed in the liquid waste management system. The system comprises collection tanks, process pumps and vessels, monitor tanks, and appropriate instrumentation and controls. The principal waste treatment process is ion exchange.

The gaseous waste management system collects, processes, and monitors gaseous releases. The process gas subsystem receives fission gases and uses charcoal delay beds to retain and delay the process gases for decay before release. The process vent subsystem collects low-activity aerated gas streams from the potentially contaminated vent header and releases the gaseous effluents through monitored filtered paths to the common plant vent.

Solidification, packaging, and storage of radioactive wastes before shipment to offsite burial grounds are controlled by the solid waste management system (SWMS) that handles radioactive wastes generated during station operation.

The process and effluent radiological monitoring and sampling system encompasses the detection and measurement of radioactive materials in plant process and effluent streams.

This evaluation is based on the staff's review of CESSAR-DC Chapter 11, which includes applicable portions of ABB-CE's January 24, 1992, response to staff's request for additional information (RAI) and ABB-CE's response to combined license (COL), confirmatory and open items identified in Chapter 11 of the draft safety evaluation report (DSER) of September 1992. The staff used the acceptance criteria in NUREG-0800, Standard Review Plan (SRP) Sections 11.1 through 11.5 as the basis for its evaluation. The subject SRP sections include compliance with 10 CFR Part 50, Appendix I and applicable general design criteria (GDC) as acceptance criteria. Additionally, in lieu of 10 CFR 20.106, which the subject sections include as an acceptance criterion, the staff used compliance with Section 20.1302, which is the current requirement. For evaluating the source terms, the staff reviewed the adequacy of information in the CESSAR-DC against the requirements spelled out in 10 CFR Sections 50.34 and 50.34a.

The staff evaluated the capability of the liquid and gaseous radwaste management systems to keep radioactive effluents as low as is reasonably achievable in accordance with 10 CFR Part 50, Appendix I requirements, based on radwastes expected to be produced over the life of the plant. Additionally, the staff evaluated the capability of the systems to maintain releases below the limits of 10 CFR Part 20, Section 1302, during periods of fission-product release, and the system's abilities to meet processing demands during normal operation, including anticipated operational occurrences (AOOs). The staff also evaluated the quality group and seismic design classification applied to equipment and structures housing these systems, compliance of design features with GDC 3 for protecting gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen, GDC 60 for controlling radioactive releases to unrestricted areas, and GDC 61 for assuring adequate safety under normal and accident conditions. Further, the staff evaluated the waste gas processing system's compliance with Branch Technical Position Effluent Treatment Systems Branch (BTP ETSB) 11-5, "Postulated Radioactive Releases Due to Waste Gas System Leak or Failure."

In its evaluation of the solid radwaste management system, the staff considered design objectives, including volumes and activities of wastes processed for offsite shipment, conformance to Federal packaging regulations, provisions for controlling potentially radioactive airborne dusts generated during compacting operations, and provisions for onsite storage before shipping.

Because specific compliance with 10 CFR Part 50, Appendix I, and the guidelines in American National Standards Institute (ANSI) N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities;" Regulatory Guide (RG) 1.21, "Measuring and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants;" and RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operation)-Effluent Streams and the Environment," are not totally within the scope of the CESSAR-DC, the staff will review COL applications on a plant-specific basis, to ensure their conformance with 10 CFR Part 50, Appendix I, ANSI N13.1, and RGs 1.21 and 4.15. Therefore, the subject conformance was identified as COL Action Item 11.1-1 in the DSER. CESSAR-DC Section 11.1 (Amendment Q) calls for the COL applicant to ensure the above conformance. Therefore, COL Item 11.1-1 is acceptable.

The staff calculated the expected releases of radioactive materials via liquid and gaseous effluents using the pressurized-water reactor (PWR) GALE code methodology described in NUREG-0017, "Calculation of Releases of

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Radioactive Materials in Liquid and Gaseous Effluents From Pressurized Water Reactors," Revision 1, April 1985. The calculations in the code for estimating the liquid and gaseous effluents during normal plant operation, including AOOs, are based on: (1) data from operating reactors; (2) field and laboratory tests; (3) standardized primary coolant activities and adjustment factors derived from the American Nuclear Society 18.1 Working Group recommendations; (4) standardized secondary coolant activities for a PWR with U-tube SGs derived from the above groups' recommendations; (5) release and transport mechanisms that result in the appearance of radioactive material in liquid streams; and (6) the plant's radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environs. The principal parameters used in these calculations, based on the tables given in CESSAR-DC Sections 11.1, 11.2, and 11.3 are listed in Table 11.1 of this report. Based on these calculations, the staff finds the liquid and gaseous effluents given in CESSAR-DC Tables 11.2-1 and 11.3-4, acceptable.

Before the DSER was issued, the staff found that the earlier version of CESSAR-DC and the ABB-CE's response to RAI gave the staff useful information for evaluating the releases of radioactive material via liquid and gaseous effluents during normal plant operation, including AOOs. However, at that time, ABB-CE failed to provide a few details (e.g., cation bed flowrate on an average basis; liquid radwaste adjustment to account for anticipated occurrences such as operator errors resulting in unplanned releases; demonstration of compliance with applicable 10 CFR Part 20 concentration limits for radioactive materials in liquid and gaseous effluents based on one-percent failed fuel of fission products; explanation of the choice of 0.03 m³/min (1 ft³/min) for flow rate of carrier gas (charcoal delay-bed system); effective decontamination factor for each waste kind; release point description for gaseous effluents) to enable the staff to evaluate independently the system's compliance with 10 CFR Part 20, 10 CFR 50.34a, and 10 CFR Part 50, Appendix I and applicable GDC. Therefore, the staff identified the lack of the above information as DSER Open Item 11.1-1.

Also, at that time, ABB-CE indicated that the radwaste systems would be designed on 0.25-percent failed fuel (ABB-CE's response to Q410.136d and Q410.137b) and had used the same basis for determining compliance with the applicable 10 CFR Part 20 concentration limits for radionuclides in liquid and gaseous effluents. SRP Sections 11.2 and 11.3 state that the radwaste systems should have the capability to process wastes based on one-percent failed fuel. The failed fuel percentage used by ABB-CE was identified as DSER Open Item 11.1-2. Subsequently, by amendments up to and including Amendment U to

CESSAR-DC Sections 11.2 and 11.3, ABB-CE provided the requested details. The staff has reviewed the details and finds them acceptable because these details comply with the acceptance criteria of SRP Sections 11.2 and 11.3 (for specific information on the subject compliance of the liquid and gaseous waste management systems, see Sections 11.2 and 11.3 of this report). Therefore, DSER Open Items 11.1-1 and 11.1-2 are resolved.

The staff noted a discrepancy in the earlier version of CESSAR-DC regarding the value for the shim bleed rate. Different subsections and tables in Section 11.2 of the document gave different values. However, the staff found that ABB-CE calculated equilibrium primary coolant concentrations correctly using the bleed rate and not the environmental discharge rate. So, the DSER called for ABB-CE to correct the discrepancy and identified it as Confirmatory Item 11.1-1. Subsequently, by amending CESSAR-DC Section 11.2, ABB-CE resolved the discrepancy. Therefore, DSER Confirmatory Item 11.1-1 is resolved.

On this basis, the staff concludes that ABB-CE has submitted sufficient information with regard to source terms in accordance with requirements of 10 CFR 50.34 and 50.34a. The staff concludes that CESSAR-DC radwaste source terms for evaluating radwaste management systems meet the acceptable criteria of SRP Section 11.1, as discussed in Sections 11.2 and 11.3 of this report, and are, therefore, acceptable.

11.2 Liquid Waste Management System

11.2.1 System Description and Review Discussion

The liquid waste (radwaste) management system (LWMS) consists of process equipment and instrumentation necessary to collect, process, monitor, and recycle or discharge the processed liquid radwastes. The staff used the acceptance criteria given in SRP Section 11.2, which are system compliance with 10 CFR Sections 34.a and 20.1302 (Section 20.1302 in lieu of Section 20.106), 10 CFR Part 50, Appendix A, GDC 60 and 61, and 10 CFR Part 50, Appendix I (for information on these requirements, see Section 11.1 of this report) as its basis for evaluating the LWMS. The LWMS comprises the following subsystems: (1) high-level waste subsystem; (2) low-level waste subsystem; (3) laundry and hot shower (LHS)/chemical waste subsystem; and (4) containment cooler condensate subsystem. The high-level waste subsystem processes the boron-containing wastes from the chemical and volume control system (CVCS) and reactor coolant system (RCS), namely, the shim bleed and equipment drains from the reactor and equipment drain tanks. Additionally, it processes the clean wastes which consist of reactor-grade laboratory drains and

Table 11.1 Principal parameters used in the calculation of gaseous and liquid effluents for CESSAR-DC

<u>Parameter</u>	<u>Value</u>
Thermal Power, MWt	3931
Total Steam Flow Rate, kg/h	8.0E6 (1.764E7 lb/h)
Primary Coolant Mass, kg	3.0E5 (6.643E5 lb)
Letdown Purification Rate, L/minute	272.5 (72 gpm)
Number of U Tube Steam Generators (SG)	2
Mass of Water in each SG, kg	9.675E4 (2.133E5 lb)
SG Blowdown Rate, kg/h	1.6E4 (3.53E4 lb/h)
Con. Demin. Regeneration Time, days	21.6
Feedwater Fraction Through Con. Demin.	0.65
Steam/Water Concentration:	
Halogens	0.01
Particulates	0.005

Liquid Waste InputsShim Bleed Waste*

Waste Collection Rate, m ³ /day	7.3 (1931 gpd)
DF for Halogens; Cs and Rb; Others**	1E4; 2E3; 1E4
Collection Time; Process Time, days	80; 0.315
Fraction Discharged	0.1

Equipment Drain Waste*

Waste Collection Rate, m ³ /day	0.946 (250 gpd)
PCA Fraction	1.0
DF for Halogens; Cs and Rb; Others	1E4; 2E3; 1E4
Collection Time; Process Time, days	80; 0.315
Fraction Discharged	0.1

Clean Waste

Waste Collection Rate, m ³ /day	2.65 (700 gpd)
PCA Fraction	0.2
DF for Halogens; Cs and Rb; Others	1E3; 1E3; 1E3
Collection Time; Process Time, days	30; 0.315
Fraction Discharged	0.1

Dirty Waste

Waste Collection Rate, m ³ /day	12.1 (3200 gpd)
PCA Fraction	0.021
DF for Halogens; Cs and Rb; Others	1E3; 1E3; 1E3
Collection Time; Process Time, days	8.5; 0.315
Fraction Discharged	1.0

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<u>Parameter</u>	<u>Value</u>
<u>Regeneration Waste</u>	
Waste Collection Rate, m ³ /day	148.7 (39300 gpd)
DF for Halogens; Cs and Rb; Others	1; 1; 1
Collection Time; Process Time, days	2.34, 0.256
Fraction Discharged	1.0
<u>Chemical/LHS Waste</u>	
DF for Cs, Others	10, 1
<u>Gaseous Waste Inputs</u>	
Full Letdown Flow Gas Stripping	Continuous
Holdup Time for re and kr Stripped from the Primary System, days	30, 3
Waste Gas System-Particulate Filter Efficiency, %	99
Auxiliary (Radwaste/Nuclear Annex/ Subsphere) Bldg. - Iodine and Particulate Filter Efficiencies, %	90, 99
Fuel Bldg. - Iodine and Particulate Filter Efficiencies, %	95, 99
<u>Containment Inputs for Gaseous Waste</u>	
Free Volume, m ³	9.46E4 (3.34E6 cf)
Internal Cleanup-Iodine and Particulate Filter Efficiencies, %	90, 99
Internal Cleanup Flow Rate, m ³ /min	1700 (60000 cfm)
Low-Volume Purge-Total Time in a Year, %	1.0
Low-Volume Purge Flow Rate, m ³ /min	35.4 (1250 cfm)
Low-Volume Purge-Iodine and Particulate Filter Efficiencies, %	70, 99
Number of High-Volume Purges per year	2
High-Volume Purge-Iodine and Particulate Filter Efficiencies, %	95, 99

Note

*10 percent of the shim bleed and equipment drain wastes collected are processed by the LWMS. The remaining 90 percent are processed by the boric acid concentrator (see Section 11.2.1 of this report for further information).

**Excludes dissolved noble gases and tritium.

aerated equipment drains. The low-level waste subsystem processes the dirty wastes which come from the plant floor drains, fuel pool liner leakage, steam generator (SG) drains and equipment, and area non-detergent decontamination water. Occasionally, containment cooler condensate tank contents also may be processed by the low-level waste subsystem. The LHS/chemical waste subsystem processes the chemical wastes (i.e., decontamination wastes, chemical samples, and chemical lab drains) and the LHS (detergent) wastes. The containment cooler condensate subsystem collects the containment cooler condensate in one of the two containment cooler condensate tanks, each of capacity 16.1 m³ (4,252 gal) and, generally, discharges the tank contents to the industrial waste discharge, since normally, the contents have low radioactivity. However, if radioactivity is detected when the contents are sampled, the contents are dispatched to the low-level waste subsystem to be processed along with other low-level wastes described above. SG blowdown is processed by the SG blowdown treatment system, and the processed blowdown is normally totally recycled in the plant. Therefore, in the System 80+ design, the SG blowdown normally does not contribute to any liquid radwaste discharge to the environs.

The LWMS comprises collection tanks, process pumps and vessels, monitor or sample tanks, and appropriate instrumentation and controls to permit most of the process to be conducted remotely. All waste collection and waste monitor tanks are equipped with continuous level indication with provisions for high-level alarm both locally and in the radwaste control room. Each type of liquid waste is segregated to minimize the potential for mixing with other types, and contaminating non-radioactive streams. Each of the high- and low-level waste subsystems contains two waste collection tanks, and two waste monitor tanks, each of capacity 128.7 m³ (34,000 gal). The LHS/chemical waste subsystem has two chemical waste collection tanks and two chemical waste sample tanks, each of capacity 36 m³ (9,600 gal) and two LHS collection tanks and two LHS sample tanks, each of capacity 25 m³ (6,700 gal). The collection and monitor tanks for the high-level, low-level, and LHS/chemical waste subsystems are provided in pairs so that one will be available to receive waste while the other is being processed or discharged. Each waste stream has a centrifugal pump that can be cross-connected to another pump in case of pump failure.

The principal process equipment for treating liquid radwaste is a set of non-regenerative ion exchangers or demineralizers. Each of the three major wastes, namely, the high-level, low-level and LHS/chemical waste is processed by its respective processing equipment. The processing equipment for the three subsystems are identical and each consists of at least one waste process filter per collection tank downstream of the tank, one carbon bed

filter (to enhance process filtration), and four ion exchangers all in series. The waste process filters are bag-type polyethylene filters and they are skid mounted in groups of two or four. The four ion exchangers are made up of one specific ion demineralizer (e.g., demineralizer loaded with Zeolite or other sluicible media to enhance ion exchange adsorption for cesium and rubidium), one cation bed, and two mixed bed demineralizers. The system piping permits connection of mobile processing equipment.

Additional details on management of the high-level, low-level and LHS/chemical wastes are given below.

The shim bleed and equipment drains from the reactor and equipment drain tanks are first collected in a 1,646 m³ (435,000 gal) capacity holdup tank after being processed by a pre-holdup mixed bed ion exchanger. Approximately 90 percent of the tank inventory is processed by a boric acid concentrator and the distillate is cycled back to the RCS. The concentrates end up in the boric acid storage tank. The balance of 10 percent is dispatched to one of the two high-level waste collection tanks mentioned above. The waste is processed by the high-level waste processing equipment described above. The processed waste is collected in one of the two high-level waste monitor tanks, sampled, and discharged if found acceptable for discharge. The staff estimates that approximately 10 percent of the shim bleed and equipment drain wastes that are generated, normally will be processed by the high-level waste processing equipment and then discharged to the environment. The staff agrees with ABB-CE that the normal generation rate of the shim bleed and equipment drain wastes requiring processing will be approximately 0.8 m³/day (218 gpd). The capacity of the limiting processing equipment in this system is 327 m³/day (86,400 gpd). This provides adequate margin for processing a surge in the shim bleed and equipment drain waste generation rate.

As stated above, clean wastes are also processed by the high-level waste subsystem. The wastes are collected in one of the two high-level waste collection tanks, processed by the subsystem processing equipment, discharged and subsequently collected in one of the two subsystem monitor tanks, and discharged (if found acceptable for discharge), after sampling. The staff estimates that approximately 10 percent of the processed clean wastes will be discharged to the environs and the balance will be recycled in the plant. The staff agrees with ABB-CE that the normal generation rate of the clean wastes will be approximately 2.7 m³/day (700 gpd). The capacity of the limiting processing equipment is 327 m³/day (86,400 gpd) as stated above. The combined normal generation rate of the shim bleed, equipment drains, and clean waste that require processing by the high level waste processing equipment is

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3.5 m³/day (918 gpd). Considering the capacity of the limiting processing equipment in the subsystem, the staff finds that there is adequate margin for processing a surge in the normal generation rates of these wastes.

Dirty wastes identified above are collected in one of the two low-level waste collection tanks, processed by the subsystem processing equipment, subsequently collected in one of the two subsystem monitor tanks, and discharged (if found acceptable for discharge), after sampling. The staff estimates that all of the processed dirty waste will be discharged to the environs. The staff agrees with ABB-CE that the normal generation of dirty wastes will be approximately 12.1 m³/day (3,200 gpd). The capacity of the limiting processing equipment is 327 m³/day (86,400 gpd) in this subsystem. This provides adequate margin for processing a surge in the dirty waste generation rate.

Chemical wastes described above are collected in one of the two chemical waste collection tanks, and LHS wastes are collected in one of the two LHS collection tanks. Normally, these wastes are discharged to the environs without any processing, if such discharge is found acceptable after sampling. However, if required, these wastes can be processed by one of the two respective tank filters and subsequently by the subsystem carbon bed filter and set of four demineralizers in series. The processed chemical waste is collected in one of the two chemical sample tanks, and the processed LHS waste is collected in one of the two LHS sample tanks. The tank contents are sampled and discharged to the environs (if found acceptable for discharge). The staff estimates the detergent and decontamination wastes to be 2 m³/day (540 gpd) and ABB-CE estimates these wastes to be 3 m³/day (800 gpd). In estimating the releases from this subsystem, ABB-CE has assumed no credit for processing the wastes, except for cesium radionuclides in the waste stream prior to their release. The tanks provided in this subsystem ensure adequate margin to collect any surge in the generation of chemical and LHS wastes. The capacity of the limiting processing equipment is 327 m³/day (86,400 gpd) in this subsystem. This provides an adequate margin for processing any surge in the LHS/chemical waste generation rate.

In addition to these LWMS subsystems, liquid radwaste can also be released to the environs from discharge of condensate cleanup system polisher regenerant solution. The System 80+ design uses regenerative condensate polishers in the condensate cleanup system (see CESSAR-DC Section 10.4.6 and Section 10.4.6 of this report). Regenerant wastes that come from the operation of the condensate cleanup system are collected in one of the two 435 m³ (115,000 gal) capacity regenerant solution neutralization (collection) tanks located in the turbine building and then discharged to the environs without

processing (if found acceptable for such discharge) after sampling the tank contents. However, if processing is required, the wastes are dispatched to the low-level waste subsystem for processing and discharge to the environs. ABB-CE states that all of the condensate cleanup system polisher regenerant wastes will be discharged to the environs. ABB-CE estimates the waste generation to be approximately 149 m³/day (39,300 gpd), and the staff agrees with this conservative estimate. The capacity of the equipment for discharging this waste (the discharge pump) is 1,363 m³/day (360,000 gpd). The collection tank capacity and the discharge pump capacity assure a reasonable margin for collecting and discharging a surge in the conservatively estimated generation rate for this waste. In estimating the liquid radwaste release to the environs from the condensate cleanup system waste, ABB-CE has assumed no credit for processing the waste prior to its release to the environs.

As discussed above, liquid waste processed prior to release, as applicable, is released to the environs from any monitor, sample, or collection tank only after sampling the subject tank contents shows that such a release is permissible. The LWMS and the regenerant waste system intermittently discharge liquid effluent in batches to the environment. All LWMS discharges of detectable radioactivity are made through a single discharge line to the environs. The condensate cleanup system polisher regenerant waste is discharged through another discharge line to the environs. All liquid waste discharges are diluted by a dilution flow of at least 2.8 m³/sec (100 ft³/sec). All releases to the environs are monitored prior to dilution and discharge by two radiation monitors, one on the common discharge line for the LWMS discharges and the other on the discharge line for the condensate cleanup system polisher regenerant waste discharge. These monitors are located downstream of the waste monitor or sample tanks for the LWMS and downstream of the regenerant waste collection tank for the regenerant wastes. Each of these radiation monitors will terminate liquid waste releases to unrestricted area before the discharge concentration via the associated discharge line exceeds a pre-determined setpoint for the applicable monitor, to comply with 10 CFR Part 20, Appendix B, Table 2, Column 2 limits for liquid effluent concentrations of radionuclides in unrestricted areas. To ensure such compliance, CESSAR-DC Section 11.2.5 calls for the COL applicant to provide the operational setpoint for the radiation monitors in the plant-specific offsite dose calculation manual (ODCM). ABB-CE has further indicated that the operational setpoints should ensure that the sum of the ratios of instantaneous discharge concentrations of radionuclides in an unrestricted area to the liquid effluent concentration limits given for the corresponding radionuclides in the table column referenced above does

not exceed 10. The staff finds this approach acceptable for demonstrating compliance with the subject regulation. As discussed above, the radiation monitors provide for control and monitored release of liquid radwastes to unrestricted areas as required by GDC 60 and 64.

ABB-CE has calculated the annual liquid effluent releases (CESSAR-DC Table 11.2-1) using PWR-GALE code methodology. The standard design parameters for running the subject computer program to calculate expected primary and secondary coolant radionuclide concentrations and liquid effluents are given in CESSAR-DC Tables 11.1.1-1, 11.2-2, and 11.2-6. Table 11.2-6 lists such LWMS equipment as number of collection and monitor or sample tanks, and filters and demineralizers and their design parameters, such as capacity and flowrates. The parameters in the other tables include (but are not limited to) collection rates, effective primary coolant activity (PCA) fractions, collection and process times, effective process decontamination factors (DFs) for different categories of radionuclides, and discharge fractions for all the waste streams (shim bleed, equipment drains, clean wastes, dirty wastes, chemical and LHS wastes, and regenerant wastes). Staff's review of the subject tables, Section 11.2 and Figures 11.2-1 and 11.2-2 of CESSAR-DC shows that the primary and secondary coolant radionuclide concentrations given in CESSAR-DC Table 11.1.1-2 are in accordance with GALE code methodology.

In reviewing ABB-CE's response to DSER Open Item 11.1-1, the staff also finds that the design does not have a cation bed in the primary purification system to remove preferentially cesium and rubidium radionuclides from the primary system. The staff finds this acceptable since there is no regulatory requirement that a cation bed should be provided. Further, the staff finds that other parameters including the DFs for the shim bleed stream and process times have been properly calculated. Additionally, to the total radionuclide release to the environment via liquid effluent obtained from running the GALE code, ABB-CE has conservatively added the contribution due to an assumed release of 189 m³/yr (50,000 gal) to the environment of secondary coolant at secondary coolant radionuclide concentration levels. For these reasons, the staff finds the radionuclide releases given in CESSAR-DC Table 11.2-1, which include liquid radwaste adjustment quantity of 5,920 MBq/yr (0.16 Ci/yr) to account for such anticipated occurrences as operator errors resulting in unplanned releases, acceptable. Therefore, the staff agrees with ABB-CE that the total liquid effluent during normal plant operation, including AOOs, will not exceed 34,780 MBq/yr (0.94 Ci/yr) for radionuclides other than tritium and 1.52×10^7 MBq/yr (410 Ci/yr) for tritium for the System 80+ design. This finding, in conjunction with the

ABB-CE assumed minimum dilution flow of 2.8 m³/sec (100 ft³/sec), provides reasonable assurance that the LWMS for the System 80+ design will meet applicable Appendix I dose guidelines for liquid effluents. However, since demonstration of specific compliance with Appendix I dose guidelines are not within the scope of the standard design, the staff will review such compliance demonstration on a plant-specific basis for each COL applicant. As stated in Section 11.1 of this report, CESSAR-DC has identified a COL action item in this regard, which the staff finds acceptable.

Using primary coolant concentrations corresponding to one-percent failed fuel or expected values (as given by GALE code run), whichever is greater for fission products, and expected values for corrosion products, annual liquid effluent data as given by GALE code run, and minimum liquid effluent dilution flow of 2.8 m³/sec, ABB-CE has demonstrated that the sum of the ratios of the annual average liquid effluent concentrations of radionuclides in any unrestricted area to the liquid effluent concentration limits for the respective radionuclides given in 10 CFR Part 20, Appendix B, Table 2, Column 2, is well below 1.0 (CESSAR-DC Table 11.2-5 value is 0.091). The staff has independently verified that the annual average liquid effluent concentrations in an unrestricted area for the System 80+ plant will be well below the limits specified in the subject table column. Therefore, the staff finds that the System 80+ design will comply with 10 CFR Part 20, Section 20.1302, since the subject section requires that the annual average concentrations of radioactive materials in effluents in an unrestricted area do not exceed the limits specified in the subject table column.

The LWMS is a non-safety-related system and it has no accident mitigation functions. The system is located in the radwaste building. The system and the structure housing the system, are designed in accordance with the applicable guidelines of Positions C.1, C.4, C.5, and C.6 of RG 1.143, "Design Guidance for Radwaste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," with respect to the specific guidelines for liquid radwaste systems design, construction and testing criteria for radwaste management systems in general, seismic design criteria for the system and the structure housing the LWMS and quality assurance (QA) for radwaste management systems. CESSAR-DC Section 11.2.1.2 details how the design of the system and its housing structure comply with the applicable guidelines of RG 1.143. Specifically, the subject section states that the LWMS is designed and tested to the codes and standards listed in Table 1 supplemented by guidelines 1.1.2 and 1.1.4 of RG 1.143. It further states that the foundations and walls of structures housing the LWMS are designed in accordance with Position C.5 (seismic design

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criteria) of the guide to a height sufficient to contain the maximum liquid inventory expected to be in the radwaste building. It also states that all tanks inside and outside have level indicators; all tank overflows, drains, and sample lines are routed to the LWMS for collection and processing; all indoor tanks have curbs or elevated thresholds with floor drains routed to the LWMS for collection and processing; and outdoor tanks, such as the boric acid storage tank have a seismic retention pond or dike to prevent an uncontrolled release of radioactive liquid to the environs should the tanks overflow or fail. Additionally, the section states that systems with potential for containing radioactive liquids are segregated from nonradioactive systems to minimize the potential for the spread of contamination into unmonitored and non-radioactive systems and ductwork in the area. The subject section states that the QA program for the installation, procurement and fabrication of the system components will comply with the Position C.6 of RG 1.143, and that the COL applicant will demonstrate such compliance. On the basis of this information and the staff's review of CESSAR-DC Section 11.2, particularly, Section 11.2.1.2, the staff finds that ABB-CE has addressed all aspects of RG 1.143 satisfactorily and that the LWMS complies with the applicable guidelines of RG 1.143, and thus complies with GDC 60 and 61 and 10 CFR 50.34a insofar as they relate to control of radioactive materials released to the environment via liquid effluents assuring adequate safety of equipment that may contain liquid radwastes under normal and postulated accident conditions, and adequacy of design information for the LWMS, respectively.

ABB-CE has also reviewed the applicability of IE Bulletin (IEB) 80-05, "Vacuum Conditions Resulting in Damage to Chemical Volume Control System (CVCS) Holdup Tanks (Sometimes Called 'Clean Waste Receiver Tanks')" to the System 80+ standard design. The subject bulletin addresses the issue concerning the release of radioactive material or other adverse effects as a result of low-vacuum conditions causing tank buckling. The low-vacuum condition is created by cooling hot water in a low pressure tank. On the basis of its review, ABB-CE has included its findings in CESSAR-DC Sections 11.2.2.2.1 and 11.2.2.2.2. The subject sections state that all tanks have either: (1) tank vents sized adequately to prevent the collapse of the tank during draindown or (2) a vacuum breaker, as necessary. The staff finds these features adequately address the concern identified in IEB 80-05 and are, therefore, acceptable. These design features comply with GDC 61, insofar as they assure adequate safety for the subject tanks under normal and postulated accident conditions.

After reviewing the version of CESSAR-DC Section 11.2 that preceded the issuance of DSER, the staff identified DSER Confirmatory Item 11.2-1. The subject item called

for ABB-CE to incorporate its January 24, 1992, response to the staff's RAI relating to the LWMS for the System 80+ standard design, in amended CESSAR-DC Section 11.2. The staff has reviewed CESSAR-DC and finds that ABB-CE has incorporated in Section 11.2 that part of the response that continues to be valid currently. Therefore, DSER Confirmatory Item 11.2-1 is resolved.

In the earlier version of CESSAR-DC Section 11.2, the staff found that ABB-CE had demonstrated compliance of liquid radwaste discharge concentrations in unrestricted areas with 10 CFR Part 20, Appendix B, Table ii, Column 2 maximum permissible concentrations (MPC), based on 0.25-percent failed fuel instead of one-percent failed fuel; the staff found no indication that the liquid radwaste adjustment quantity of 5920 MBq/yr (0.16 Ci/yr) to account for anticipated occurrences had been included in assessing the total annual liquid radwaste release to the environs, and the staff found the quoted DFs for the shim bleed stream was significantly higher than the DFs for other streams. For these reasons, the staff designated the above concerns as DSER Open Items 11.2-1 and 11.2-2. In response to these concerns, CESSAR-DC Section 11.2 has been amended. As discussed above, the amended Section 11.2 demonstrates compliance with 10 CFR Part 20, Appendix B, Table 2, Column 2 (which lists the liquid effluent concentration limits for radionuclides at the boundary of the unrestricted area), based on one-percent failed fuel and also includes the liquid radwaste adjustment quantity in the calculated total liquid effluent. Further, the amended Table 11.2-2 and Figure 11.2-2 have deleted the originally included credit due to CVCS purification demineralizer in determining the effective DF for processing the shim bleed stream by the high-level waste processing equipment. The credit for the CVCS purification demineralizer has, however, been included in determining the equilibrium primary concentrations of radionuclides, which is proper. On this basis, DSER Open Items 11.2-1 and 11.2-2 are resolved.

In the earlier version of CESSAR-DC Section 11.2, the staff did not find an adequate discussion on how the design meets all the guidelines of RG 1.143, including Position C.6 that addresses the QA aspects of radwaste management systems. Therefore, the staff designated this lack of information as DSER Open Item 11.2-3. As discussed above, the staff finds that all aspects of RG 1.143 have been satisfactorily addressed in amended CESSAR-DC. On this basis, DSER Open Item 11.2-3 is resolved. In the earlier version of CESSAR-DC Section 11.2, the staff found no information on the disposition of shim bleed and clean wastes that are not discharged to the environs. Therefore, the staff designated this issue as DSER Open Item 11.2-4. Amended Figure 11.2-2 shows the disposition (for additional information see the review discussion

on shim bleed and clean wastes given above). On this basis, DSER Open Item 11.2-4 is resolved.

11.2.2 Conclusion

In its evaluation of the LWMS, the staff considered: (1) the capability of the system to maintain concentrations of radionuclides in liquid effluents in unrestricted areas below their respective liquid effluent concentration limits referred to in 10 CFR Part 20, Section 20.1302, during periods of fission-product leakage at design levels from the fuel (i.e., one percent); (2) the capability of the system to meet the processing demands of the station during AOOs; (3) the quality group and seismic design classification applied to the equipment, components, and structures housing the system; (4) the design features that are incorporated to comply with GDC 60 as it relates to control of radioactive materials released to the environment via liquid effluents; and (5) the design features to comply with GDC 61 as it relates to assuring adequate safety for the equipment in the LWMS under normal and postulated accident conditions. The staff reviewed all applicable information provided in CESSAR-DC Section 11.2, and the ABB-CE submittal dated January 24, 1992, in response to the staff's RAI.

On the basis of its review, as discussed in Section 11.2.1 of this report, the staff concludes that ABB-CE has submitted sufficient design information for the System 80+ LWMS in accordance with 10 CFR 50.34a requirements. Additionally, the staff concludes that the CESSAR-DC LWMS includes the equipment and design features necessary to control the releases to unrestricted areas of radioactive materials in liquid effluents, in accordance with GDC 60, and assure adequate safety under normal and postulated accident conditions in accordance with GDC 61. The staff further concludes that the LWMS meets the requirements of 10 CFR Part 20, Section 20.1302 with respect to liquid effluent concentration limits in unrestricted areas. On this basis, the staff concludes that the System 80+ LWMS meets the applicable acceptance criteria of SRP Section 11.2 and is, therefore, acceptable.

11.3 Gaseous Waste Management System

11.3.1 System Description and Review Discussion

The gaseous waste management system (GWMS), and the plant ventilation exhaust systems control, collect, process, store, and dispose of gaseous radioactive wastes generated during normal operation, including AOOs. The staff has used the acceptance criteria given in SRP Section 11.3 which are system compliance with 10 CFR Part 50, Section 50.34a and 10 CFR Part 20, Section 20.1302 (Section 20.1302 in lieu of 20.106), 10 CFR Part 50, Appendix A, GDC 3, 60, and 61, and 10 CFR Part 50,

Appendix I (for information on these requirements, see Section 11.1 of this report) for its evaluation of the GWMS.

The GWMS comprises two subsystems: the process gas subsystem and the process vent subsystem. In the writeup that follows, the staff includes in the GWMS, not only the management of process gas and process vent (described below), but also the management of building ventilation, containment purge, and condenser evacuation system exhausts in so far as they relate to gaseous effluents to environs. The process gas portion of the GWMS receives fission gases from the process gas header (PGH) and uses charcoal adsorber beds to retain and delay the process gases for decay before release to the plant vent to the environs via the nuclear annex ventilation exhaust. The primary input sources to the PGH are gases stripped continuously from the RCS by the CVCS gas stripper, and the volume control tank. The reactor drain tank and equipment drain tank process vent surge volumes constitute the balance of the hydrogenated gaseous inputs to the PGH. The flow in the PGH primarily consists of hydrogen and noble gases with some trace quantities of other fission gases and water vapor. The continuous removal of fission gases from the RCS by the gas stripper keeps the RCS fission gas concentrations at a low level. The process gas subsystem uses charcoal at ambient temperature to delay the passage of radioactive gases through the system. The process gas subsystem includes a charcoal guard bed and six charcoal adsorber beds. A condenser cools the stripped gases and condenses and removes water vapor to a dew point below 7 °C (45 °F) before the gas enters the charcoal adsorber beds. Furthermore, downstream of the cooler condenser, the system has a humidity analyzer which detects inadequate moisture removal. The design data for the process gas subsystem, including the design pressure of the charcoal beds and cooler condenser, are given in CESSAR-DC Table 11.3-7.

The process gas subsystem has radiation monitors which monitor the discharge from the charcoal beds upstream of the discharge to the nuclear annex ventilation system. The discharge is automatically terminated when the radiation level in the discharge reaches a pre-determined setpoint set by the COL applicant for the radiation monitor. Thus, the system is consistent with GDC 60 with regard to control of radioactive release to unrestricted areas.

In its calculation of gaseous effluent from the process gas subsystem, ABB-CE used delay times of 30 days and 3 days for xenon and krypton radionuclides, respectively. Using a charcoal mass of 6,940 kg (15,300 lb) in the adsorber beds and a carrier gas flowrate of 0.028 m³/min (1 scfm) through the adsorber beds, ABB-CE calculated the above delay times. Since ABB-CE offered no basis for

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the chosen flow rate in the earlier version of CESSAR-DC Section 11.3, the staff designated the lack of basis for the chosen flow rate as DSER Open Item 11.3-1. In a letter dated November 4, 1993, ABB-CE stated that the chosen value was based on Chapter 12 specifications for PWRs given in the Electric Power Research Institute Utility Requirements Document for advanced light water reactors. ABB-CE further stated that operating experience at PWRs that use charcoal delay beds for waste gas treatment (e.g., Surry and Beaver Valley units) showed less than 0.0028 m³/m (0.1 scfm) average flow rate through the delay beds, and that the higher chosen value was therefore conservative, since delay time varies inversely as the flow rate. On the basis of this satisfactory response, DSER Open Item 11.3-1 is resolved. The staff also finds that the ABB-CE calculated delay times are in agreement with NUREG-0017, Revision 1, methodology. The process gas subsystem effluent is additionally processed, if required, by the nuclear annex filtration system, which consists of a pre-filter, high-efficiency particulate air (HEPA) filters, and a charcoal adsorber before it is discharged to the plant vent via the nuclear annex ventilation system. Thus, by including charcoal delay beds, the process gas subsystem is consistent with GDC 60 with regard to control of radioactive release to unrestricted areas.

The GWMS is a non-safety-related system and has no accident mitigation functions. The process gas portion of the system is located in the nuclear annex, which is a seismic Category I structure and, therefore, designed to withstand a safe-shutdown earthquake (SSE). The process gas subsystem and the structure housing the subsystem are designed in accordance with the applicable Positions C.2, C.4, C.5, and C.6 of RG 1.143 with respect to specific guidelines for gaseous radwaste systems; general guidelines for design, construction, and testing criteria for radwaste systems; specific seismic design criteria for gaseous waste management system; general seismic design criteria for structures housing radwaste systems; and general guidelines for providing QA for radwaste management systems. CESSAR-DC Subsection 11.3.1.2 provides a detailed discussion of how the design of the subsystem and its housing structure meet the applicable guidelines of RG 1.143. Specifically, the subject section states that the GWMS (i.e., process gas portion) is designed and tested to the codes and standards listed in Table 1 of RG 1.143 supplemented by Guidelines 2.1.2 and 2.1.4 of RG 1.143. It further states that the foundations and walls of the structures housing the charcoal adsorber beds and the supports for the beds are designed to meet the seismic design criteria specified in Position C.5 of RG 1.143. The subject section states that the QA program for the installation, procurement, and fabrication of the system components will comply with the Position C.6 of RG 1.143 and that the COL applicant will demonstrate such compliance.

The earlier version of CESSAR-DC Section 11.3 did not address the QA compliance.

On the basis of the above information and its review of amended CESSAR-DC Section 11.3, particularly Subsection 11.3.1.2, the staff finds that ABB-CE has addressed all aspects of RG 1.143 satisfactorily and that the GWMS complies with the applicable guidelines of RG 1.143, and thus complies with GDC 60 and 61 and 10 CFR 50.34a, insofar as they relate to control of radioactive materials released to the environment via gaseous effluents, assuring adequate safety under normal and postulated accident conditions for the equipment that may contain gaseous radwastes, and adequacy of design information for the GWMS, respectively. On this basis, DSER Open Item 11.3-2 (lack of sufficient details in the earlier version of CESSAR-DC, Section 11.3 to demonstrate GWMS compliance with applicable guidelines of RG 1.143) is resolved.

Since the potential exists for buildup of explosive mixtures of hydrogen and oxygen in the process gas subsystem, the subsystem has to be either designed to withstand the effects of a hydrogen explosion, or must have design features to preclude the formation or buildup of explosive mixtures. The earlier (i.e., prior to the issuance of System 80+ DSER) version of CESSAR-DC addressed the issue by identifying the design provisions to prevent the formation or buildup of explosive mixtures in the GWMS. Specifically, the earlier version stated that the system would include gas analyzers with alarm capability to detect the formation of explosive gas mixtures, and that nitrogen would be added when required to dilute the gas mixture and thus prevent it from becoming explosive. However, the earlier version did not give sufficient details on the remedial actions that would have to be performed, the automatic control function of the analyzers, control settings of the alarms, and the details of the nitrogen system, for the staff to conclude that the system will meet GDC 3 with respect to protection of the system from the effects of hydrogen explosion. Therefore, the staff designated the lack of details pertaining to the protection of the GWMS against possible hydrogen explosion as DSER Open Item 11.3-3. Subsequently, ABB-CE amended CESSAR-DC Section 11.3, in response to this concern. As amended, Section 11.3 states that the charcoal vessels, condenser cooler, piping, components, and valves within the GWMS will be designed to withstand a hydrogen explosion (i.e., the system design pressure will be 20 times the normal operating pressure of the system as stated in SRP Section 11.3, Acceptance Criterion B.6.a). Furthermore, one oxygen and one hydrogen analyzer will be installed in the system and an alarm will be annunciated in the control room when the oxygen concentration in the system reaches one percent by volume.

Section 11.3 also states that such remedial actions as manually eliminating the source of oxygen entry into the system and/or adding nitrogen diluent will be performed, upon receipt of the high alarm at one-percent oxygen. The analyzers will be within the detonation resistant pressure boundary of the system. As amended, CESSAR-DC Section 11.3 addresses the staff's concerns identified above satisfactorily, and demonstrates that the process gas portion of the GWMS will comply with GDC 3 as it relates to protecting the subject portion of the GWMS from the effects of an explosive mixture of hydrogen and oxygen. On this basis, DSER Open Item 11.3-3 is resolved.

The process vent portion of the GWMS collects low-activity aerated gas streams from the potentially contaminated vent headers in the nuclear annex and radwaste building. The process vents, except those from the condenser evacuation system, are monitored, filtered as required through the nuclear annex ventilation filters, and released through the common plant vent. The condenser evacuation system exhaust is monitored and discharged through the common plant vent. Other gaseous streams exhausted to the common plant vent through HEPA and charcoal adsorber filters are: the containment high- and low-purge exhausts and the exhausts of the plant ventilation systems for the fuel building and the auxiliary building (i.e., the reactor subsphere, the nuclear annex, and radwaste building). ABB-CE has submitted a simplified diagram of all gaseous effluents in CESSAR-DC Figure 11.3-2 which shows that the ventilation exhausts from the fuel building, radwaste building, and nuclear annex will be filtered, as required. The turbine building exhaust will not be filtered or monitored since it is not expected to have detectable radioactivity. It is released to the environs via turbine building vents.

In the earlier version of CESSAR-DC Section 9.4, ABB-CE described the exhaust filtration systems for the plant ventilation systems identified above, except for the radwaste building ventilation system. The filtration system includes a prefilter, an upstream HEPA filter, a charcoal adsorber, and a downstream HEPA filter. However, from the earlier version of CESSAR-DC Chapters 9 and 11, it was not clear whether the radwaste building exhaust filtration system would have all the above components, since CESSAR-DC Chapter 11 indicated that the subject system would have all the components whereas Section 9.4 did not have such information. Therefore, the staff designated this as DSER Confirmatory Item 11.3-2. Amended CESSAR-DC Section 9.4 includes all the components in the radwaste building exhaust filtration system. On this basis, DSER Confirmatory Item 11.3-2 is resolved.

CESSAR-DC Section 9.4 and Table 11.3-2 state that the design and testing of the ventilation exhaust filtration systems will meet the guidelines of RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," or RG 1.52, "Design, Testing and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," whichever is appropriate. The subject table gives the design data on the ventilation exhaust filtration systems and containment internal cleanup filtration system. RG 1.52 is referred to above, since the exhausts from containment high purge, reactor subsphere and fuel building during normal operation including AOOs will be through a filtration system, designed and tested in accordance with RG 1.52 guidelines. Furthermore, CESSAR-DC includes Tables 9.4-5 and 9.4-6 which demonstrate compliance of engineered-safety- feature (ESF) grade and non-ESF grade filtration systems with RG 1.52 and RG 1.140 guidelines, respectively. On this basis, the ventilation exhaust treatment systems, where provided, comply with the guidelines of RG 1.52 or RG 1.140, whichever is appropriate. Therefore, the systems comply with GDC 60 and 61, as they relate to control of the release of radioactive material from plant areas, including the fuel storage and handling area, to the environs.

ABB-CE has calculated the annual gaseous effluent releases (CESSAR-DC Table 11.3-4) using PWR-PALE code methodology. The standard design parameters for running the subject computer program are given in CESSAR-DC Tables 11.1.1-1, 11.2-2, and 11.3-2, and Sections 11.3.2.1 and 11.3.6.1. Staff review of the subject tables and sections shows that the expected primary and secondary coolant radionuclide concentrations given in CESSAR-DC Table 11.1.1-2 are in accordance with PALE code methodology. Further, the staff finds that filter efficiencies for removal of radioiodine and particulates from effluent streams from different building ventilation systems, and containment internal cleanup filter efficiencies used in the PALE run, are in accordance with RGs 1.52 or 1.140 guidelines, whichever is appropriate. Also, the staff finds that the dynamic adsorption coefficients for krypton and xenon in the delay beds used in the PALE run have been conservatively calculated. These, in conjunction with a conservatively assumed carrier gas flow rate, result in conservatively calculated delay times for krypton and xenon in the charcoal delay beds. For these reasons, the staff finds acceptable the radionuclide releases to the environs via gaseous effluents during normal plant operation, including AOOs, given in CESSAR-DC Table 11.3-4. Therefore, the staff agrees with ABB-CE that the expected gaseous effluent will not exceed 8.65×10^7 MBq/yr (2,337 Ci/yr) for noble gases, 2.66×10^9

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MBq/yr (0.072 Ci/yr) for iodines, 2.63×10^1 MBq/yr (7.1×10^{-4} Ci/yr) for particulates, 4.44×10^7 MBq/yr (1,200 Ci/yr) for tritium, 1.26×10^6 MBq/yr (34 Ci/yr) for Argon-41 and 2.7×10^5 MBq/yr (7.3 Ci/yr) for Carbon-14. All these releases will be via the monitored unit vent. Since the ingestion, inhalation, and ground contamination doses to applicable organs of an offsite individual resulting from release of radioiodines, radioactive material in particulate form, tritium and Carbon-14 via airborne effluents depend upon a number of site-specific parameters and population exposures and associated cost-benefit analysis also depend upon site specific parameters, the staff has limited its evaluation of CESSAR-DC compliance with Appendix I guidelines to GWMS compliance for external doses (due to noble gas radionuclides in gaseous effluents) in unrestricted areas. Gaseous effluent data given in CESSAR-DC Table 11.3-4 in conjunction with ABB-CE assumed atmospheric dispersion factor (χ/Q) of 7.2×10^{-5} sec/m³ and RG 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I," Revision 1, dose factors for noble gases assure that GWMS for System 80+ design will meet Appendix I dose guidelines for external doses from gaseous effluents in unrestricted areas for sites that have χ/Q equal to or less than the above quoted value. For the assumed χ/Q , the doses will be well within the applicable Appendix I dose guidelines as shown in CESSAR-DC Table 11.3-5. However, since demonstration of specific compliance with Appendix I dose guidelines are not within the scope of the standard design, the staff will review such compliance demonstration on a plant-specific basis for each COL application. As stated in Section 11.1 of this report, ABB-CE has identified COL Action Item 11.1-1 in this regard, which the staff finds acceptable.

The earlier version of CESSAR-DC Table 11.3-5, compared the radionuclide concentrations in gaseous effluents to unrestricted areas with the applicable 10 CFR Part 20 MPC. The table showed that the sum of the fractions of radionuclide concentrations in effluents to unrestricted areas to their respective MPCs was a very small fraction (0.0013) of the allowable total (i.e., 1.0). In the DSER, the staff identified the following concerns relating to the analysis:

- (1) The conversion factor should be 3.17×10^{-8} instead of 3.17×10^{-11} .
The conversion factor represents the multiplication factor to be applied on the product of the annual release of radionuclides in Ci/yr via gaseous effluents and atmospheric dispersion factor in sec/m³ to obtain the gaseous effluent radionuclide concentrations in $\mu\text{Ci/cc}$.

- (2) The atmospheric dispersion factor 4.97×10^{-4} sec/m³ is inappropriate and also contradicts the long-term annual average value of 7.2×10^{-5} sec/m³ given in CESSAR-DC Section 11.3.6.3.
- (3) The SRP Section 11.3 guidelines state that for PWRs, demonstration of compliance of radionuclide concentrations in gaseous effluents in unrestricted areas, with 10 CFR Part 20 gaseous effluent concentration limits in unrestricted areas, should be based on design basis source terms, i.e., one percent failed fuel for fission products. Therefore, the multiplication factor of seven applied to the gaseous effluent values listed in CESSAR-DC Table 11.3-4 to obtain the gaseous effluent values corresponding to the design basis source terms is incorrect, since the Table 11.3-4 gaseous effluent values do not correspond to 0.14 percent failed fuel for all fission products. This is because the Table 11.3-4 effluent values are based on System 80+ primary coolant concentrations of radionuclides under normal operating reactor conditions (obtained by running the PWR GALE code with System 80+ parameter inputs) which, in turn, are based on measured primary coolant radionuclide concentrations for PWRs and adjustments for System 80+ specific parameters.
- (4) The values in CESSAR Table 11.3-4 are partly based on a carrier gas flow rate of 0.03 m³/min (1 scfm) through the delay beds and ABB-CE has not explained the basis for this flow rate.

In the DSER, the staff stated that ABB-CE needed to resolve all the items listed above, revise the analysis and table, and demonstrate that the total fraction is less than 1.0 for the staff to conclude the GWMS design is in compliance with 10 CFR 20.106. The subject regulation requires that the annual average radionuclide concentrations in gaseous effluents at the boundary of the unrestricted area do not exceed their respective unrestricted area MPC listed in 10 CFR Part 20, Sections 20.1 through 20.602, Appendix B, Table ii, Column 1 (these sections and this table are in the 1993 version of 10 CFR, but these have been deleted in the 1994 version of 10 CFR). Therefore, the staff designated its concerns as DSER Open Item 11.3-4. By Amendment U to CESSAR-DC Section 11.3, ABB-CE responded to these concerns by revising the analysis to demonstrate GWMS design compliance with 10 CFR 20.1302 (which is mandatory effective January 1, 1994), in lieu of 10 CFR 20.106. Specifically, using primary coolant concentrations corresponding to one-percent failed fuel, or expected primary coolant concentrations (as given by GALE code run) whichever is greater, for fission products, and expected primary coolant concen-

trations for corrosion products, annual gaseous effluent data as given by GALE code run, long-term annual average atmospheric dispersion factor of 7.2×10^{-5} sec/m³, same carrier gas flow rate of 0.028 m³/min (acceptable to staff as discussed above) through the charcoal delay beds and correct conversion factor of 3.17×10^8 , ABB-CE has demonstrated that the annual average gaseous effluent concentration of radionuclides in exclusion area boundary (EAB) (CESSAR-DC Table 11.3-6) will be below the gaseous effluent concentration limits listed in 10 CFR Part 20, Appendix B, Table 2, Column 1. Table 11.3-6 shows that the sum of the ratios of annual average gaseous effluent concentrations of radionuclides at the EAB to their respective concentration limits given in the subject 10 CFR Part 20 table is less than 1. For these reasons, the staff considers that the DSER Open Item 11.3-4 is resolved and that the System 80+ GWMS design complies with 10 CFR 20.1302.

ABB-CE included a process gas subsystem failure analysis in the earlier version of CESSAR-DC to demonstrate that the system design meets the guidelines of BTP ETSB 11-5. The subject BTP stipulates that the total body dose at the EAB due to release of radioactivity for 2 hours resulting from postulated failure of the process gas subsystem, calculated in accordance with the BTP assumptions, should not exceed 5 mSv (500 mrem). The staff noted that in the analysis, ABB-CE had multiplied the gaseous effluents (calculated using NUREG-0017, Rev. 1 methodology) by seven to obtain the design-basis source terms for use in the analysis. The earlier analysis was not conservative since the gaseous effluents calculated using NUREG-0017, Revision 1, methodology does not correspond to 0.14-percent failed fuel for all noble gases. Since the BTP explicitly states that the source term should correspond to one-percent failed fuel, the staff considered that ABB-CE should submit a revised analysis. Therefore, the staff designated the need for reanalysis of the process gas subsystem failure as DSER Open Item 11.3-5.

By Amendment V to CESSAR-DC Section 11.3, ABB-CE responded to the above concern of the staff by revising the analysis to demonstrate the process gas subsystem design compliance with the subject BTP's dose criterion mentioned above. Using a short term (0-2 hour) χ/Q of 1.0×10^{-3} sec/m³ and other assumptions in accordance with the BTP, the subject section has calculated a 0-2 hour total body dose at the EAB, which is well within 5 mSv (500 mrem). The staff has independently analyzed the above failure and has determined that the 0-2 hour total body dose at the EAB will be well within 5 mSv (500 mrem). Therefore, DSER Open Item 11.3-5 is resolved.

After reviewing the version of CESSAR-DC Section 11.3 that existed prior to the issuance of the DSER, the staff

identified DSER Confirmatory Item 11.3-1. The subject item called for ABB-CE to incorporate its January 24, 1992, response to staff's RAI relating to the GWMS for the CESSAR-DC standard design, in amended System 80+ Section 11.3. The staff has reviewed CESSAR-DC and finds that ABB-CE has incorporated in Section 11.3 that part of the response that continues to be valid currently. Therefore, DSER Confirmatory Item 11.3-1 is resolved.

11.3.2 Conclusion

In its evaluation of the GWMS design, the staff considered: (1) capability of the system to maintain gaseous effluents in unrestricted areas below the limits in 10 CFR Part 20, Section 20.1302, during periods of fission product leakage at design levels from the fuel (i.e., one percent); (2) the capability of the system to meet the processing demands of the station during AOOs; (3) the quality group and seismic classification applied to the equipment, components and structures housing the system; (4) the design features incorporated to control the releases of radioactive materials to environs via gaseous effluents in accordance with GDC 60; (5) the design features incorporated to comply with GDC 61, insofar as it relates to assuring adequate safety under normal and postulated accident conditions for the equipment that may contain gaseous radwastes; (6) the design features incorporated to comply with GDC 3 as it relates to protecting the process gas portion of the GWMS from the effects of an explosive mixture of hydrogen and oxygen; and (7) the design features to ensure the process gas subsystem design compliance with BTP ETSB 11.5 guidelines. The staff reviewed all applicable information submitted in CESSAR-DC Section 11.3.

On the basis of this review, as discussed in Section 11.3.1 of this report, the staff concludes that the System 80+ GWMS complies with 10 CFR 20.1302, 10 CFR 50.34a, and GDC 3, 60, and 61. Therefore, the staff concludes that System 80+ GWMS design meets the acceptance criteria of SRP Section 11.3 and is, therefore, acceptable.

11.4 Solid Waste Management System

11.4.1 System Description and Review Discussion

The SWMS consists of equipment and instrumentation to collect, segregate, store, process, sample, and monitor solid wastes. The staff has used the acceptance criteria given in SRP Section 11.4, which are system compliance with 10 CFR 50.34a as it relates to providing adequate system design information, 10 CFR 20.1302 (in lieu of Section 20.106 which is deleted in the 1994 version of 10 CFR) as it relates to ensuring concentrations of radionuclides in gaseous and liquid effluents to unrestricted areas

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arising from SWMS operation within the limits specified in 10 CFR Part 20, Appendix B, Table 2, Columns 1 and 2, respectively, 10 CFR Part 50, Appendix A, GDC 60 and 64 as they relate to controlling and monitoring the release of radioactive materials to the environment, GDC 63 as it relates to monitoring radiation levels and leakage, 10 CFR Part 71 as it relates to packaging of radioactive materials, and 10 CFR Part 61 as it relates to classifying, processing, and disposing solid wastes, as the basis for its evaluation of the SWMS design. The SWMS processes both wet solid wastes (e.g., spent resins, filter cartridges and bag filters used in the LWMS), and dry active wastes (e.g., rags, paper, clothing) for shipment to a licensed burial site.

The wet solid wastes consist of high-activity spent resins from fuel pool demineralizers, and other demineralizers used to process primary reactor coolant, such as, the purification and pre-holdup ion exchangers in the CVCS, high-activity cartridge filters, low-activity spent resins from the LWMS demineralizers, low-activity LWMS bag filters, and low-activity resins from the secondary side regenerant condensate polishers and SG blowdown demineralizers. The dry solid wastes consist of heating, ventilation, and air conditioning (HVAC) system air cleanup system filters and compactible wastes such as rags, contaminated clothing, and paper.

One spent resin tank of volume 13.6 m³ (3,600 gal) and two spent resin tanks each of 13.6 m³ (3,600 gal), collect and hold high and low activity spent resins, respectively. Additionally, there are two spent resin surge tanks, each of capacity 5.3 m³ (1,400 gal), one each for the low and high activity spent resins. These surge tanks provide surge space in the otherwise closed loop resin transfer systems for the low and high activity spent resins. Each spent resin tank has a stainless steel spent resin transfer pump which transfers the spent resins from the various demineralizers to the spent resin tank. Spent resin filters are provided on each spent resin transfer pump discharge line. Each spent resin tank also has a stainless steel positive displacement spent resin forwarding pump to forward resin slurries to the wet solid radwaste dewatering processing area. In the dewatering processing area, the spent resins are dewatered and stored in high integrity containers (HICs) for eventual shipment to a licensed burial facility. The radwaste building includes a solidification processing area, and should solidification of the spent resins before packaging them in shipping containers be required by the burial facility, the spent resins will be solidified in the solidification processing area using an approved solidification agent. Filled shipping containers are stored in a shielded storage area of the radwaste building until they are shipped.

When the high-activity cartridge filter (which is on the discharge line of the high-activity spent resin transfer pump) needs replacement, it is removed by a remote handling device from the filter housing and put in a shielded container, as necessary, after the housing contents are flushed to the equipment drain tank, if required, and the housing is vented and drained. Subsequently, the cartridge is transported and lowered into a shipping container (HIC or 0.21 m³ (55 gal. drum)) in a filter bunker. After sufficient decay, the container is removed from the bunker and transported to a shielded storage space in the radwaste building for eventual shipment to a licensed burial facility. The low-activity bag filters (which are on the discharge lines of low-activity spent resin transfer pumps) that need replacement are lifted from their housing and placed in adjacent shielding containers (usually HICs, although 0.21 m³ (55 gal.) drums may be used) by remote handling tools, after water is purged from the filter housing and the filter medium is dewatered using compressed process air. The filled containers are stored in a shielded storage area for eventual shipment to a licensed burial facility.

As stated in Section 11.2 of this report, regenerant polishers will be used in the condensate cleanup system. The secondary side resins from these polishers will be processed as necessary and packaged for disposal if the resins become physically broken or the DF is reduced. At this time, the resins will be sluiced into the shipping container and dewatered in the turbine building. HICs will be used only as necessary to ensure compliance with the Department of Transportation (DOT) regulations. The SG blowdown treatment demineralizer resins will also be processed in the turbine building similar to the condensate cleanup system resins; however, these resins will not be regenerated. A spent resin decanting tank of sufficient capacity is installed in the turbine building to facilitate processing of the secondary side resins, as necessary, based on sampling of the tank contents.

Dry solid wastes, such as contaminated cloth, paper, and plastic are compacted by a dry solid compactor in the low-level waste handling and packaging area. These and non-compatible dry wastes, including HVAC system filter assemblies are packaged in 0.21 m³ (55 gal. drums) and stored in a low-level solid waste storage area of the radwaste building for eventual shipment.

The spent resin tanks and the shipping containers are sampled and surveyed to verify that the dewatering or solidification (if required) and packaging are complete and meet the guidelines of BTP ETSB 11-3, which provides the design guidance for SWMS. Additionally, the wet solid wastes will be processed and disposed in accordance with 10 CFR Part 61 requirements with regard to waste classifi-

cation (Section 61.55 gives the criteria for determining whether the wastes are Class A, B, or C and the criteria which the different waste classes should satisfy for disposal) and waste characteristics (Section 61.56 spells out the various characteristics the processed wet wastes should satisfy). To ensure the above compliance, ABB-CE has identified a COL activity (CESSAR-DC Section 11.4.1.1, Item F) which calls for the owner operator to process and classify the packaged waste in accordance with 10 CFR Part 61 requirements. Specifically, the COL activity calls for the owner operator to develop operating procedures (i.e., boundary conditions for a set of process parameters such as settling time, drain time, drying time, etc.) for processing the wet solid wastes to assure that 10 CFR Part 61 requirements are met. The staff recognizes that such development of procedures (i.e., process control program (PCP)) to ensure that the SWMS will operate as intended is within the scope of the COL applicant. The staff will, therefore, review the PCP, including dewatering or solidification (if performed), on a plant-specific basis against BTP ETSB 11-3 guidelines and 10 CFR Part 61 requirements. Since ABB-CE did not identify this COL responsibility in the earlier version of CESSAR-DC Section 11.4, the staff designated the lack of such identification for ensuring compliance with 10 CFR Part 61 requirements, as DSER COL Action Item 11.4-1. CESSAR-DC Section 11.4 has been subsequently amended to include the subject COL action item as discussed above. Therefore, COL Action Item 11.4-1 is acceptable. In this context, the staff notes that CESSAR-DC Section 11.4.1.1 states that all solid waste material will be packaged in accordance with 10 CFR Part 71 and DOT regulations prior to their shipment to a licensed burial site and that the COL applicant will develop procedures to comply with these regulations. The staff finds this acceptable since it recognizes developing packaging procedures is within the scope of the COL applicant.

The liquid and gaseous effluents resulting from the SWMS operation are released during normal operation, and during AOOs to unrestricted areas through the LWMS and the radwaste building ventilation system, respectively. The SWMS is designed so that liquids removed during the dewatering process of wet solid waste are routed back to the LWMS to be processed before release to the environment. Liquids released from a SWMS component failure are collected and routed to the LWMS for processing. In addition, curbing surrounds the major components, such as the resin storage tanks, to contain their contents in the event of a failure. The resin-storage tanks and shipping containers have non-clogging wire screens to prevent inadvertent discharge of resin beads to the environment. The gases collected in the dry active storage waste processing area are discharged through the radiation building ventilation system to the plant vent. Gases resulting from

other operations of the SWMS are also discharged through the radiation building ventilation system to the plant vent. The dry solids compactor has an air filtration system which includes a HEPA filter. A fan draws air through the HEPA filter and the exhaust gases generated by compaction are also passed through the plant vent via the radiation building ventilation system, preventing an unfiltered release of airborne contamination to the environment. There are area radiation monitors in the radwaste building areas, radwaste building loading bay, and solid waste drum storage area. The liquids and gases that result from the SWMS operation are monitored before the release to environs by the LWMS and GWMS monitors. As concluded in Sections 11.2 and 11.3 of this chapter, the liquid and gaseous effluents comply with 10 CFR 20.1302. On the basis of the above discussion, the staff finds that the SWMS complies with GDC 63, 60, and 64 with respect to controlling monitoring leakages, and controlling and monitoring releases of radioactive materials to the environment, respectively. Also, the staff concludes that the system complies with 10 CFR 20.1302.

The SWMS is a non-safety-related system and has no accident mitigation functions. The system is located in the radwaste building, which is a seismic Category II building; however, it is designed for the SSE using seismic Category I criteria. The system, and the structure housing the system, are designed in accordance with the applicable guidelines C.3, C.4, C.5, and C.6 of RG 1.143 with respect to specific guidelines for solid radwaste systems; general guidelines for design, construction, and testing criteria for radwaste systems; general seismic design criteria for structures housing radwaste systems; and general guidelines for providing QA for radwaste management systems. CESSAR-DC Section 11.4.1.2 provides a detailed discussion of how the design of the system and its housing structure are consistent with the applicable guidelines of RG 1.143. Specifically, the subject section states that the SWMS is designed and tested to the codes and standards listed in Table 2 of RG 1.143, supplemented by Guidelines 3.1.2 and 3.1.4 of RG 1.143. It further states that the foundations and adjacent walls of the structures housing the SWMS are designed to meet the requirements specified in Regulatory Position 5 of RG 1.143 to a height sufficient to contain the maximum liquid inventory in the building. The subject section states that an analysis will be performed to demonstrate the compliance with the above guideline. As stated above, curbing surrounds such major components as resin-storage tanks to contain their contents in the event of a failure. Accident releases due to a major component failure or SWMS leak will be contained in the radwaste building. Section 11.4.1.2 additionally states that the QA program for the installation, procurement and fabrication of the system components will comply with the Regulatory Position C.6 of RG 1.143 and that the COL

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applicant will develop a QA program for construction and operations to demonstrate such compliance. On the basis of this information and the staff's review of CESSAR-DC, particularly, Section 11.4.1.2, the staff finds that ABB-CE has addressed all aspects of RG 1.143 satisfactorily, and that the SWMS meets the applicable guidelines of RG 1.143, and thus complies with GDC 60, 63, and 10 CFR 50.34a. In the earlier version of CESSAR-DC, Section 11.4 did not address the QA compliance. This finding by the staff also resolves the DSER Open Item 11.4-1 (lack of sufficient details in the earlier version of CESSAR-DC Section 11.4 to demonstrate SWMS compliance with guidelines of RG 1.143) identified in the DSER.

On the basis of annual reports of radioactive materials released from nuclear power plants (NUREG/CR-2907, "Radioactive Materials Released From Nuclear Power Plants," Volumes 9 and 11 for the years 1988 and 1990 - only PWRs were considered), the staff estimates the processed wet wastes (i.e., spent resins and filter cartridges and low-activity bag type filters) that will have to be shipped annually from the System 80+ PWR to be about 47 m³ (1,700 ft³) containing approximately 1.26 x 10⁷ MBq (340 Ci). ABB-CE, based on 4-year annual shipment data for seven nuclear units, estimates this to be 55.2 m³ (1,950 ft³) containing approximately 1.2 x 10⁷ MBq (320 Ci). ABB-CE estimates about 25 percent of this volume will be made up of high-activity spent resins which, in turn, includes (but is not limited to) primary reactor coolant cleanup spent resins. The remaining 75 percent will be made up of low-activity spent resins from LWMS demineralizers and secondary side condensate and steam generator blowdown cleanup resins. On the basis of these reports, the staff estimates the processed dry wastes that will have to be shipped annually from the System 80+ PWR to be about 96 m³ (3,400 ft³) containing approximately 4.8 x 10⁵ MBq (13 Ci). ABB-CE estimates this to be about 58 m³ (2,060 cf) containing approximately 5.6 x 10⁵ MBq (15 Ci). In the earlier CESSAR-DC version of Section 11.4, ABB-CE gave a low dry waste volume and did not explain how that volume was determined. Further, the earlier version did not give the curie content of both the dry and wet wastes that will require to be shipped annually to a licensed burial facility. The staff, therefore, designated these concerns as DSER Open Item 11.4-2. Subsequently, by Amendment 0 to Section 11.4 of CESSAR-DC, ABB-CE has revised the dry and wet waste volumes it expects to be shipped annually and their curie content. These are given above. The subject section further has explained the basis for the values, i.e., has provided operating data for seven nuclear units for 4 recent years. ABB-CE estimates are in reasonable agreement with staff's independent estimates as shown above. Therefore, DSER Open Item 11.4-2 is resolved.

CESSAR-DC Section 11.4 states that the radwaste building includes adequate space for processing the dry and wet wastes, as well as, storing the processed wet and dry wastes that can be expected to be generated annually at normal generation rate (given above). Specifically, the subject section and CESSAR-DC Figure 11.4-1 show the various storage areas, some of which are shielded (to store HICs, high-activity cartridge filters, etc.) and which include (but are not limited to) sorting and staging space to separate contaminated from uncontaminated material. On the basis of this discussion, and the staff's estimate of processed waste volumes that can be expected to be shipped annually (given above), the staff finds that the available storage space is sufficient to accommodate one full offsite waste shipment of dry wastes and 30 days of wet waste generation at normal generation rate in accordance with BTP ET'SB 11-3, Positions B.III.2 and 3. On the basis of the capacities of the spent resin storage tanks in the radwaste building (given above), and the expected annual generation of spent resin, the staff finds that the capacities of the tanks accumulating spent resins are consistent with BTP Position B.II.1 (storage of primary system spent resins generated in 60 days and other spent resins generated in 30 days).

In Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," the staff provided guidance to licensees on the addition of onsite storage facilities for low-level radioactive wastes generated onsite. The staff recognizes that the need for additional onsite storage capacity for low-level radioactive wastes beyond what has been provided for in the System 80+ standard design is a site-specific issue, since it will depend upon offsite low-level waste storage space availability for the site's wastes. Therefore, when such a need is identified by a COL applicant, and the COL applicant submits the details of its proposed onsite low-level radioactive waste storage facility to the NRC, the staff will review and evaluate such a proposed site-specific facility against the guidelines in GL 81-38. The staff notes that the guidance in the subject GL is similar to the guidance in Appendix 11.4-A to SRP Section 11.4.

As discussed above, because of surge tanks, and the capacity to store processed wastes beyond the one month requirement, the staff finds that the SWMS is designed with sufficient storage to accommodate surges in normal waste generation. The radwaste building has additional space for placement of leased temporary equipment to accommodate modifications to the SWMS as new processes and configurations become available. The SWMS also contains connections for use of vendor supplied services such as rapid dewatering or waste drying systems when it is determined that the use of these methods represents cost savings over permanently installed alternatives.

In the DSER, the staff requested ABB-CE to incorporate its January 24, 1992, response to the staff's RAI relating to the SWMS for the System 80+ standard design, in amended CESSAR-DC Section 11.4. This was identified as DSER Confirmatory Item 11.4-1. The staff has reviewed the amended CESSAR-DC and finds that ABB-CE has incorporated, in Section 11.4 that part of the response that continues to be currently valid. Therefore, DSER Confirmatory Item 11.4-1 is resolved.

11.4.2 Conclusion

In its evaluation of the SWMS, the staff considered: (1) the capability of the system in conjunction with LWMS and GWMS to maintain any liquid or gaseous effluents in unrestricted areas, arising from the system operation, below the limits in 10 CFR 20.1302; (2) the capability of the system to meet the processing demands of the station during AOOs; (3) system design objectives in terms of expected types, volumes, and activities of wastes processed for offsite shipment; (4) quality group and seismic classification applied to the structures housing the system; (5) provisions for onsite storage of processed solid wastes before shipment; and (6) design features to comply with GDC 60, 63, and 64. The staff reviewed all the information provided in CESSAR-DC Section 11.4. CESSAR-DC Section 11.4 information also includes COL action items for compliance with 10 CFR Parts 61 and 71.

On the basis of this information, as discussed above in Section 11.4.1 of this report, the staff concludes that the System 80+ SWMS design complies with 10 CFR 20.1302, and GDC 60, 63, and 64. The staff further concludes that CESSAR-DC Section 11.4 contains sufficient design information in accordance with 10 CFR 50.34a requirements. Additionally, the staff concludes that the SWMS design will comply with 10 CFR Parts 61 and 71 requirements, and DOT regulations; however, since demonstration of such compliance falls within the scope of the COL applicant, the staff will review compliance of SWMS design with 10 CFR Parts 61 and 71 on a plant-specific basis for each COL application. For these reasons, the staff concludes that the System 80+ SWMS design conforms with the acceptance criteria of SRP Section 11.4 and is, therefore, acceptable.

11.5 Process and Effluent Radiological Monitoring and Sampling System

11.5.1 System Description and Review Discussion

The process and effluent radiological monitoring and sampling system is used to measure, record, and control releases of radioactive materials in plant process streams and effluent streams. The system consists of permanently

installed sampling and monitoring equipment designed to indicate routine operational radiation releases, equipment or component failure, system malfunction or misoperation, and potential radiological hazards to plant personnel or to the general public.

The staff has reviewed ABB-CE's response of January 24, 1992, to the staff's RAI of October 10, 1991, relating to the system; ABB-CE's submittal dated November 4, 1993, relating to the system; and amended CESSAR-DC Section 11.5. The staff has used the acceptance criteria given in SRP Section 11.5, which are system compliance with 10 CFR 20.1302 (in lieu of 10 CFR 20.106 which has been deleted in 1994 version of 10 CFR) as it relates to ensuring concentrations of radionuclides in gaseous and liquid effluents to unrestricted areas within the limits specified in 10 CFR Part 20, Appendix B, Table 2, Columns 1 and 2, respectively, 10 CFR Part 50, Appendix A, GDC 60 and 64 as they relate to controlling and monitoring the release of radioactive materials to the environment via gaseous and liquid effluents, and GDC 63 as it relates to monitoring fuel and waste storage areas, as the basis for its evaluation of the process and effluent radiological monitoring and sampling system. A description and discussion of the system based on the staff's review are given below.

The radiation monitoring system (RMS) is designed to provide: (1) early warning to station of personnel equipment, component, or system malfunction or misoperation, or potential radiological hazards within the station consistent with 10 CFR Part 20 and 10 CFR Part 50, Appendix I; (2) continuous monitoring of radioactive liquid and airborne releases consistent with the requirements of 10 CFR Part 20, 10 CFR Part 50, GDC 60, 63, and 64, and the guidelines of RG 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Waste and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light Water Cooled Nuclear Power Plants;" and (3) monitoring of airborne activity in selected locations and effluent paths for postulated accidents in accordance with the requirements of 10 CFR Part 50, and the guidelines of NUREG-0737, "Clarification of TMI Action Plan Requirements," and RG 1.97, "Instrumentation for LWR Nuclear Power Plants To Assess Plant and Environs Conditions During and Following Accident." The RMS monitors normal and potential paths for release of radioactive materials to give continuous indication and recording of gaseous and liquid radioactivity levels leaving the plant.

Gaseous process and effluent monitors are provided for: (1) the process gas subsystem discharge line; (2) the unit vent discharge line during normal operation; (3) the unit vent discharge line during accident situations; (4) the containment high purge exhaust and low-purge exhaust; and (5) the condenser evacuation system discharge.

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Liquid process and effluent monitors are provided for: (1) the component cooling water system discharge (two monitors); (2) the liquid waste discharge; (3) the SG blowdown sample; (4) the reactor coolant gross activity (process sampling system); (5) the turbine building drains; (6) the station service water system; (7) the containment cooler condensate tank effluent; (8) the neutralization tank (condensate polisher regenerant waste) discharge; and (9) SG blowdown system discharge.

Airborne radiation monitors are provided for: (1) the containment atmosphere; (2) the radwaste building ventilation system discharge; (3) radwaste building atmosphere (six monitors - monitored locations include HIC dewatering/washdown/inspection labelling areas, dry waste compaction area, tool/equipment decontamination area, general access areas at El 15.2 m (50 ft) and 21.3 m (70 ft) and counting room); (4) the fuel building ventilation exhaust system inlet; (5) the ventilation systems multisampler (monitors 12 locations on a rotating basis); (6) nuclear annex building ventilation exhaust; (7) the control room air intake (two monitors for each intake); (8) reactor building annulus air; (9) the reactor building subsphere ventilation exhaust; (10) the portable airborne monitor (on a mobile cart); and (11) the emergency operations facility ventilation air inlet.

The stream which is monitored, the detector type, the typical range the detector will see, the power supply to the detector, the seismic category of the detector, the automatic function associated with the monitor, if any, and the configuration/location of all the gaseous process and effluent monitors, all the liquid process and effluent monitors, and all the airborne radiation monitors are given in CESSAR-DC Tables 11.5-1, 11.5-2, and 11.5-3. The staff has reviewed these tables and finds that they cover all the applicable gaseous and liquid effluent paths, and liquid and gaseous process streams identified in SRP Section 11.5, Tables 1 and 2 and that the tables in CESSAR-DC provide the required information on the monitors. Therefore, the staff finds that these monitors comply with GDC 64 with regard to monitoring of liquid and gaseous effluents from the plant to unrestricted areas.

The area RMS monitors the radiation levels in selected areas throughout the plant. Most of these are designed to provide normal operation indication of radiological events; however, some are for post accident monitoring. A list of 18 areas with area radiation monitors is given in CESSAR-DC Table 11.5-4. The subject table gives the typical range, power source and seismic category of these monitors. These areas include (but are not limited to) new fuel storage area, spent fuel pool bridge, fuel building area, solid waste drum storage and handling area, and other radwaste building areas. These area monitors are

designed to provide normal operation indication of unusual radiological events to operators and station personnel so that they can complete corrective actions in a timely manner. Some area monitors are designed for post-accident indication of radiation levels if access to these areas for maintenance work on equipment important to safety is needed. Area radiation monitors will have local visual and audible alarms. On this basis, the staff finds that the process and effluent monitoring and sampling system for CESSAR-DC design provides the needed monitoring for fuel and radioactive waste storage and thus complies with GDC 63.

Besides the area radiation monitors in the 18 areas listed in CESSAR-DC Table 11.5-4, special-purpose area monitors are used for process monitoring functions or other special monitoring applications. These locations include: (1) the main steamlines upstream of the safety relief valves (two monitors); (2) the reactor coolant purification filters area (two monitors) (3) the RCS hot leg (two monitors); (4) the high-range containment area away from the RCS (two monitors); and (5) Nitrogen-16 steam lines (two monitors - one per SG). Typical ranges, power sources, and seismic categories for these monitors are listed in CESSAR-DC Table 11.5-5.

The RMS initiates such control actions, as reducing or terminating releases to the environment on detection of high radiation by the monitors. Specifically, the process gas subsystem discharge monitor, liquid waste discharge (from the LWMS monitor tank, detergent waste collection or sample tank, chemical waste collection or sample tank) monitor, neutralization tank discharge monitor, the containment high and low purge exhaust monitors, the turbine building drains effluent monitor, SG blowdown system discharge monitor, and containment cooler condensate tank discharge monitor initiate control actions to terminate the applicable discharge on detection of high radiation by the respective monitor. The control room air intake monitors cause the isolation of their respective air intake upon detection of high radiation level by the applicable air intake monitor. The fuel building ventilation exhaust monitor, located upstream of the fuel building exhaust filtration system, initiates control action to automatically divert the exhaust through the filtration system on detection of high radiation in the exhaust. Other radiation monitors, such as the radwaste building and nuclear annex building ventilation exhaust monitors, facilitate the manual diversion through the respective exhaust filtration system by the plant operator, if the applicable monitor detects high radiation on the applicable exhaust. The containment cooler condensate tank and neutralization tank discharge monitors facilitate manual diversion of the discharges to the low purity waste processing system by the operator, if so required. On this basis, the staff finds that the System

80+ design provides for control and monitoring of radioactivity releases to the environs in accordance with GDC 60 and 64. To comply with the numerical objectives in 10 CFR Part 50, Appendix I for the offsite doses due to gaseous and liquid effluents during normal plant operation including AOOs, plants will be required to limit annual and three-month offsite doses. Also, the plants will be required to use treatment systems (e.g., waste gas treatment by delay beds; demineralizers to treat liquid radwastes; ventilation exhaust filtration systems), if the monthly dose is likely to exceed about 25 percent of the Appendix I annual dose guidelines prorated for one month. These requirements will be specified in a plant-controlled document and will be implemented. Additionally, plants will be required to limit instantaneous discharge concentrations from the process gas subsystem and LWMS to comply with 10 CFR Part 20, Section 1302. Plants will be required to provide the associated set points for the applicable radiation monitors in the plant-specific ODCMs. These requirements, in conjunction with the automatic control (i.e., termination of the discharge) features of applicable effluent monitors, will ensure System 80+ effluent monitors are designed in compliance with 10 CFR 20.1302. The staff finds this acceptable. The staff will review the plant-specific radiological effluent technical specifications (RETS) that will be provided in the plant-controlled document as well as setpoints in the plant-specific ODCM on a plant-specific basis for each COL application.

The process and effluent radiological monitoring and sampling system has the capability to sample process and effluent streams during normal plant operation, including AOOs. These are summarized below.

The system provides for continuous representative sampling for airborne particulate and iodine radioactivity for the unit vent discharge. Additionally, the unit vent has grab sampling provisions. The major gaseous streams which have provisions for periodic grab sampling of iodine and particulate radioactivity are: (1) building ventilation system exhausts, (2) process gas subsystem exhaust, (3) main condenser evacuation system exhaust, and (4) containment purge exhausts. In CESSAR-DC Section 11.5, ABB-CE states that the sampling systems for airborne radioactivity will be designed in accordance with ANSI N13.1-1969 guidelines. The major liquid wastes that have provisions for grab sampling for gross radioactivity determination, identification, and concentration of principal radionuclides and alpha emitters are: (1) LWMS collection and monitor tanks, (2) neutralization tank, (3) component cooling water, (4) service water, and (5) SG blowdown. Effluents from LWMS collection (if applicable) and monitor tanks, neutralization tank, service water system, and component cooling water system (if applicable) are

also grab sampled for tritium. Additionally, service water system, component cooling water system, and turbine building drains have continuous sampling provisions. The sampling provisions for gaseous and liquid process and effluent streams are provided in CESSAR-DC Tables 11.5-6 and 11.5-7, respectively. On the basis of its review of these tables, the staff finds that the applicable liquid and gaseous process and effluent streams will be sampled as specified in SRP Section 11.5, Tables 1 and 2.

Periodic sampling supplements the function of the process and effluent radiation monitors. The sampling programs will be designed in accordance with RG 1.21 and the sampling requirements defined in the Technical Specifications. All continuous effluents that are potentially radioactive will be periodically sampled and analyzed. All stored wastes will be sampled, and samples will be analyzed before the wastes are released to the environment. Comparisons will be made between gross radioactivity measurements of continuous monitors and analyses of specific radionuclides as required by RG 1.21.

Because specific compliance with Appendix I to 10 CFR Part 50 and the guidelines given in ANSI N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities;" RG 1.21, "Measuring and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants;" and RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operation)--Effluent Streams and the Environment," are not within the scope of the ABB-CE System 80+ design, the staff will review COL applications to ensure their conformance with Appendix I to 10 CFR Part 50, ANSI N13.1, and RGs 1.21 and 4.15. The earlier version of CESSAR-DC, Section 11.5 did not identify any COL action item in this regard. Therefore, the staff designated demonstration of the subject conformance as COL Action Item 11.5-1 in the DSER. Amended CESSAR-DC Section 11.5.1.1 contains a COL action item which calls for the COL applicant to demonstrate the subject conformance. Therefore, COL Action Item 11.5-1 is acceptable.

Prior to the issuance of the DSER for the CESSAR-DC design, ABB-CE proposed revising the CESSAR-DC to specify that COL applicants will prepare written procedures for sample collection, preparation, and analysis in accordance with Position C of RG 4.15. ABB-CE further proposed that procedures should also be prepared for the use of radioactivity reference standards, detector calibration and checks of radiation monitor systems, and for reduction, evaluation, and reporting of data. Therefore, the staff identified this as COL Action Item 11.5-2 in the DSER. Subsequently, ABB-CE has incorporated the above proposal in CESSAR-DC Section 11.5.1.4. The subject

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section, Section 11.5.2.2, and Tables 11.5-6 and 11.5-7 further state that the frequency of sampling and other sampling procedural requirements will be specified in the applicable plant's technical specifications. On this basis, COL Action Item 11.5-2 is acceptable. The staff will review the sampling programs, including the written procedures, on a plant-specific basis against the guidelines given in ANSI N13.1-1969, RGs 4.15 and 1.21, for each COL application. The review will include (but will not be limited to) major features of the sampling program, such as sampling frequency, type of analysis that will be performed on the samples, sensitivity of analysis, and purpose of sampling. On the basis of this discussion relating to sampling of gaseous and liquid process and effluent streams, the staff finds that the CESSAR-DC Tables 11.5-6 and 11.5-7 in conjunction with COL applicant's written procedures mentioned above, and COL applicant's demonstration of compliance of monitoring and sampling programs with ANSI N13.1, RGs 1.21 and 4.15 and Appendix I, assure that all applicable (to the System 80+ standard design) liquid and gaseous process and effluent streams listed in SRP Section 11.5, Tables 1 and 2 will be sampled as specified in the tables. For this reason, the staff finds that the sampling system for the radwastes in the CESSAR-DC design complies with GDC 64, insofar as it relates to monitoring all radioactive effluent discharge paths.

In the earlier version of CESSAR-DC Section 11.5, ABB-CE did not provide sufficient information (in tabular form) on monitors and sampling provisions for liquid and gaseous process and effluent streams. Furthermore, it was not clear that all liquid and gaseous process and effluent streams listed in SRP Section 11.5, Tables 1 and 2 which are applicable to the System 80+ design, were considered with regard to monitoring and sampling provisions for the streams. Therefore, the staff designated this lack of information as DSER Open Items 11.5-1 and 11.5-2. In response to the DSER open items, ABB-CE amended CESSAR-DC Section 11.5 and provided Tables 11.5-1 through 11.5-7. On the basis of the staff's review of the amended CESSAR-DC Section 11.5 and Tables 11.5-1 through 11.5-7, as discussed above, the staff considers DSER Open Items 11.5-1 and 11.5-2 resolved.

System 80+ post-accident radiation monitors, consistent with RG 1.97, are the high-range containment area monitors, primary coolant monitors, main steamline monitors, unit vent monitor, unit vent post-accident monitor, and selected area radiation monitors which cover areas where access may be required to service equipment important to safety. The post-accident area radiation monitor locations are selected on the basis of the post-accident shielding analysis and design information about equipment location and access requirements, as discussed

in Chapter 12 of CESSAR-DC. ABB-CE states that post-accident radiation monitors meet the special requirements of RG 1.97, including equipment qualification, redundancy, power source, channel availability, QA, display and recording, range, interfaces, testing, calibration, and human factors. In its response to the staff's RAI relating to post-accident monitoring instrumentation, ABB-CE stated that these instruments are discussed in CESSAR-DC Sections 7.1.2.26 and 7.5.2.5. Also, ABB-CE submitted additional information regarding the instruments. The earlier version of CESSAR-DC Section 11.5.1.2.1 referred to Sections 7.1.2.26 and 7.5.2.5 for the additional information given in the response. Since the staff found that the earlier version of the referred sections did not convey the response fully, it recommended the incorporation of the response in CESSAR-DC Section 11.5.1.2.1 and designated the issue as DSER Confirmatory Item 11.5-1. Subsequently, ABB-CE incorporated the applicable portion of the response in CESSAR-DC Section 11.5.1.2.1. Also, ABB-CE revised Section 11.5.1.1, added a new Section 11.5.2.6, and amended Table 11.5-1. On the basis of its review of amended CESSAR-DC Section 11.5.1, the staff finds that the unit vent has a high-range noble gas monitor to monitor noble gas concentrations in effluents during and following an accident, and has the capability to continuously sample plant effluents for iodine and particulates during and following an accident. For the System 80+ standard design, only the effluent through unit vent has to be considered since all radioactive plant releases go through the unit vent. The staff finds the specific ranges for the high-range and normal range noble gas effluent monitors and containment area high-range monitor conform to the ranges required and specified for such monitors in NUREG-0737, Three Mile Island (TMI) Item II.F.1, Attachments 1 and 3. The staff further finds that ABB-CE has identified a COL action item (Subsection 11.5.2.6) which states that the COL applicant should provide an operation and maintenance manual that describes or demonstrates (as appropriate) the following: (1) procedures and/or methods for converting radiation measurement into release rate of gaseous discharge through the unit vent; (2) sampling techniques used to monitor and sample effluent gases to assure that a representative sample is taken, and that the sampling system provided is capable of maintaining isokinetic conditions during and following an accident as specified in NUREG-0737, Item II.F.1, Clarification 3 of Attachment 2; (3) collection technique used to extract a representative sample of radioactive iodine and particulates during and following an accident (the information obtained from these samples will be used to quantify the releases for dose calculations and assessments as specified in NUREG-0737, Table II.F.1-2); (4) calibration frequency and techniques for radiation monitors based on vendor information for procured equipment; and (5) design of the sampling system in compliance with

regulatory shielding requirements for low- radiation exposure during post-accident conditions specified in NUREG-0737, Item II.F.1, Clarification 2 of Attachment 2. On the basis of this information, DSER Confirmatory Item 11.5-1 is resolved.

The staff also finds that the accident monitoring instrumentation provided in the System 80+ design for monitoring noble gases in effluents, and sampling continuously and analyzing the plant effluent for post-accident releases of radioiodine and particulates in conjunction with the COL applicant's operation and maintenance manual, prepared as discussed above, will meet the requirements of NUREG-0737, TMI Item II.F.1, Attachments 1 and 2 and, therefore, will comply with the applicable portions of 10 CFR 50.34(f)(2)(xvii) (these incorporate the subject TMI requirements). The staff will review the plant-specific operation and maintenance manual discussed above on a plant-specific basis for each COL application.

The staff has reviewed the System 80+ design regarding the issues identified in NRC Bulletin 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to the Environment." Specifically, the staff has reviewed the CESSAR-DC design features (CESSAR-DC Sections 9.4, 11.2, 11.3, 11.5 and 12.3) to: (1) detect the contamination of non-radioactive systems and (2) prevent the potential for unmonitored and uncontrolled release of radioactive material to the environment. On the basis of such a review, as discussed in this section and Sections 11.2 and 11.3 of this chapter, the staff finds that in the System 80+ design, the radioactive systems are segregated from non-radioactive systems, and all radioactive or potentially radioactive effluent pathways are monitored before the effluents are released to the environment. The staff further finds that such normally non-radioactive systems as containment cooler condensate tank contents or the turbine building drains are monitored prior to their release. If found radioactive, their normal release paths are isolated and the streams are diverted to the LWMS to be processed and released to the environment. The LWMS discharges and the condensate polisher regenerant waste stream are provided with radiation monitors which automatically terminate any liquid radwaste discharge when pre-set limits are exceeded. From its review of the System 80+ design for ventilation systems (CESSAR-DC Section 9.4), the staff finds that the ventilation system air flow patterns are designed so that the air flows from areas of lower radioactivity contamination to areas of higher potential contamination to prevent the spread of contamination in usually uncontaminated areas. The radwaste building, nuclear annex, subsphere, fuel building and containment purge exhausts are released via monitored unit vent and are further filtered, as necessary, for removal of radioiodine

and particulate from the effluent streams, before release to the environs (for further information on ventilation systems, see Section 9.4 of this report). The staff also finds that CESSAR-DC Section 12.3 has additional details regarding radioactivity contamination in plant areas and source term control (for further information, see Section 12.3 of this report). On the basis of this discussion, the staff considers that CESSAR-DC Sections 9.4, 11.2, 11.3, 11.5, and 12.3 satisfactorily address the concerns raised in NRC Bulletin 80-10 and that the design features in the System 80+ design are adequate to: (1) detect the contamination of non-radioactive systems and (2) prevent the potential for unmonitored and uncontrolled release of radioactive material to the environment.

11.5.2 Conclusion

On the basis of this discussion, the staff concludes that the System 80+ process and effluent radiological monitoring and sampling system complies with 10 CFR 20.1302, and with GDC 60, 63, and 64. The staff also concludes that the system design complies with 10 CFR 50.34(f)(2)(xvii) as it relates to monitoring gaseous effluents for noble gases, and sampling and analyzing gaseous effluents for radioiodine and particulates, during and following an accident (NUREG-0737, TMI Item II.F.1, Attachments 1 and 2 requirements). Thus, the system meets the acceptance criteria of SRP Section 11.5 and is, therefore, acceptable. As stated above, the staff will review the RETS, ODCM, and the operation and maintenance manual (the subject manual for demonstrating compliance with the TMI requirements referred to above), and specific compliance with the guidelines of ANSI N13.1-1969 and RGs 1.21 and 4.15, on a plant-specific basis for each COL application.

12 RADIATION PROTECTION

Introduction

In Chapter 12 of its standard safety analysis report for design certification (CESSAR-DC) for the System 80+ plant design, ABB-CE describes the radiation protection measures of the plant design and operating policies. The staff reviewed this information against the criteria given in Chapter 12 of the Standard Review Plan (SRP) (NUREG-0800). The radiation protection measures incorporated into the System 80+ design are intended to ensure that internal and external radiation exposures to station personnel, contractors, and the general population from plant conditions (including anticipated operational occurrences (AOOs)), will be within the limits of regulatory criteria and as low as reasonably achievable (ALARA). ABB CE's radiation protection design and program features should also be consistent with the guidelines of Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable" (Rev. 3).

The basis of the staff's acceptance of the material reviewed is that doses to personnel will be maintained within the limits of 10 CFR Part 20, "Standards for Protection Against Radiation." On May 21, 1991, the Commission revised the system of radiation dose limitation in 10 CFR Part 20. The occupational dose limits for whole-body radiation exposure changed slightly from 12.5 mSv (1.25 rem) per quarter year (with a provision to extend to 120 mSv (12 rem) per year) to 50 mSv (5 rem) per year (with a provision to extend to 100 mSv (10 rem) per year). The limits in the previous 10 CFR Part 20 for doses from licensed radioactive material inside the body (deposited through injection, absorption, ingestion, or inhalation) were separate from the dose limits for exposure to licensed sources outside the body. The new Part 20 limits the sum of the external whole-body dose (deep dose equivalent) and the committed effective equivalent doses resulting from radioactive material deposited inside the body. In addition, the new Part 20 requires this sum (the total effective dose equivalent) to be maintained ALARA for each individual. These changes to the regulation do not affect the acceptance criteria used by the staff to review the System 80+ plant design. The SRP acceptance criteria ensure that the radiation doses resulting from exposure to licensed radioactive sources outside the body and inside the body can each be maintained well within the limits of 10 CFR Part 20 and the principle of ALARA. The balancing of internal and external exposure necessary to ensure that their sum is ALARA is an operational concern that will be reviewed in conjunction with a combined license (COL) application. Part 20, as amended, contains a number of new programmatic requirements that do not affect plant design. As discussed below, programmatic and operational

radiation protection concerns will be addressed by an applicant seeking a COL.

The staff has been given sufficient information by ABB-CE to conclude that the radiation protection measures incorporated in the design will offer reasonable assurance that occupational doses will be maintained ALARA and within the limits of 10 CFR Part 20 during plant operations.

The bases for the staff's conclusions follow.

12.1 Ensuring That Occupational Radiation Doses Are as Low as Is Reasonably Achievable

The staff has reviewed the information in CESSAR-DC for adherence to (1) the guidelines in RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," and (2) the criteria in SRP Section 12.1, regarding the radiation protection aspects of the System 80+ design. The staff reviewed CESSAR-DC Section 12.1 to ensure that ABB-CE had either committed to adhere to the criteria of the regulatory guides and staff positions referenced in SRP Section 12.1 or had provided acceptable alternatives. In addition, the staff selectively reviewed CESSAR-DC against acceptance criteria of the SRP, using the review procedures given there.

12.1.1 Policy Considerations

In CESSAR-DC Section 12.1.1, ABB-CE describes the design, construction, and operational policies it has implemented to ensure that the ALARA philosophy is factored into each stage of the System 80+ design process. ABB-CE commits to ensure that the System 80+ plant will be designed and constructed in a manner consistent with the guidelines of RG 8.8. This consistency will be achieved by reviewing plant design during the design phase and inspecting the shielding and piping layout during the construction phase. These policy considerations are consistent with the guidelines of RG 8.8 and are acceptable.

The amended 10 CFR Part 20 requires that all applicants develop, document, and implement a radiation protection program. This program should encompass the ALARA concept and includes provisions for maintaining radiation doses and intakes of radioactive materials ALARA. The detailed policy considerations regarding overall plant operations and implementation of such a radiation protection program are outside the scope of this review. The operational ALARA policy forms the basis for the station ALARA manual. In order to maintain doses to plant

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personnel ALARA, the COL applicant will review all plant procedures and modification plans that involve personnel radiation exposure to ensure that the ALARA philosophy is applied. The COL applicant's operational ALARA policy will conform to the requirements of 10 CFR Part 20 and the recommendations of RG 8.8, RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants" (Rev. 2), and RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposure as Low as Is Reasonably Achievable." COL applicants seeking an operating license by referencing the System 80+ certified design will be required to address these operational considerations to ensure radiation doses are ALARA. This is designated COL Action Item 12.1.1-1.

12.1.2 Design Considerations

The objectives of the general design considerations and shielding are to minimize the time employees need to spend in radiation areas and to minimize radiation levels in routinely occupied plant areas housing equipment that requires employee attention. ABB-CE states that these design considerations are consistent with the guidelines in RG 8.8 and RG 8.10.

Additional functions of the plant shielding (1) ensure that occupational radiation exposures are maintained ALARA, (2) maintain radiation exposure to control room operators within the limits of 10 CFR Part 50 (Appendix A, Criterion 19) after an accident, and (3) protect certain components from excessive activation or excessive exposure to radiation. ABB-CE states that it will not provide a detailed shielding analysis as part of the certification document since plant components have not been procured and pipe routing is not complete. The staff designated this as DSER Open Item 12.2.1-1 in the draft safety evaluation report (DSER). ABB-CE has addressed this issue (which is further discussed in Section 12.2.1 of this evaluation) in Section 3.2 ("Radiation Protection") of its Certified Design Material (CDM). Before loading fuel into the reactor, the COL applicant will submit a detailed shielding analysis. The COL applicant's radiation protection staff will assess the station design during design and construction. The COL applicant's staff will also ensure that the final design incorporates lessons learned from previous nuclear plant designs. This is designated COL Action Item 12.1.2-1.

The System 80+ design incorporates numerous design features to satisfy the design objectives of the plant radiation protection program. The use of highly reliable equipment reduces the frequency of maintenance and the associated personnel exposure. An example is the use of reliable extended service lighting in high-radiation areas to reduce the frequency of relamping. Careful attention is

paid to environmental qualification of equipment to withstand conditions such as radiation, humidity, and temperature. Electrical components containing radiation-sensitive materials are well shielded or are located in low-radiation areas. Except in cases where no proven alternative material exists, materials in contact with the reactor coolant have cobalt impurities of no more than 0.05 weight percent (wt percent) and a low nickel content, thus minimizing the production of cobalt-60 and cobalt-58, which are the major sources of radiation exposure during shutdown, maintenance, and inspection activities. Adequate spacing and laydown areas around equipment facilitates maintenance and inspection activities. A transport path and adequate rigging are furnished for removing and replacing equipment. Radioactive systems are separated from nonradioactive systems, and high-radiation sources are located in separate shielded cubicles. Equipment requiring periodic servicing or maintenance (pumps, valves, and control panels) is separated from more radioactive sources (tanks and piping). Valves located in high-radiation areas are equipped with reach rods or motor operators to minimize operator exposure. Tanks, valves, and piping are designed with smooth interior surfaces; drains are located at low points; and flushing connections minimize the buildup of crud in these components. Systems that produce radioactive waste are located close to radwaste processing systems to minimize the length of piping runs carrying highly radioactive material. This proximity also minimizes the potential for pipe plugging. These design features are consistent with the guidelines in RG 8.8 and are acceptable.

In addition to the design features just described, the System 80+ design incorporates several design features that represent an improvement over the design features used at many currently operating plants. Blanket-type thermal insulation for components carrying radioactive fluids use Velcro™ fasteners, where practical, to facilitate removal, thereby reducing personnel exposures. The plant is designed to accommodate the use of robotic technology for maintenance and surveillance in high-radiation areas. The System 80+ design minimizes the use of evaporators; these have historically needed frequent maintenance, exposing operating and maintenance personnel to substantial levels of radiation. These features to minimize personnel exposures comply with the guidelines of RG 8.8 and are acceptable.

At the time that the DSER was prepared, CESSAR-DC Sections 12.1 and 12.3 did not meet the acceptance criteria of the SRP with respect to the description of plant ALARA design features. This lack of detail was designated as DSER Open Item 12.1.2-1. In Amendments K and T to these sections of the CESSAR-DC, ABB-CE has submitted a more detailed listing and description of plant ALARA

design features that are part of the System 80+ plant design. Several of these added design features were initially described only in the System 80+ Design Certification ALARA Guideline Manual, which is not part of the Design Certification Document. With the addition of these ALARA design features to the CESSAR-DC, the staff considers DSER Open Item 12.1.2-1 to be resolved.

12.1.3 Operational Considerations

The System 80+ radiation protection program ensures that radiation exposures to employees are maintained ALARA in accordance with the requirements of 10 CFR Part 20 and the recommendations of RGs 8.8 and 8.10. Most exposure to radioactivity at operating plants occurs from maintenance and inspection activities during plant outages. Those outage activities that could involve significant radiation exposure are carefully planned by radiation protection personnel, will utilize previous operating experience, when applicable, and will be performed using appropriate exposure reduction techniques. Management will appropriately change techniques or procedures to reduce exposures during activities that require such reduction.

To reduce doses during outage activities, systems and major pieces of equipment subject to crud buildup are equipped with connections to flush and/or chemically decontaminate the system or piece of equipment to reduce the crud levels. Mockups will be used to train employees before they engage in potentially high-dose jobs. Equipment such as sound-powered telephones or closed-circuit television will be used during long-duration jobs to permit supervisors to communicate with workers from a lower radiation area and avoid exposure to an area that has higher radiation levels. Entry-exit areas are established in lower radiation areas to minimize the dose when workers don and remove protective clothing. These operational considerations comply with the guidance in RG 8.8 and are acceptable:

SRP Section 12 lists the following regulatory guides that the COL applicant should address:

RG 8.2, "Guide for Administrative Practices in Radiation Monitoring"

RG 8.3, "Film Badge Performance Criteria"

RG 8.7, "Occupational Radiation Exposure Records System"

RG 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program"

RG 8.15, "Acceptable Programs for Respiratory Protection"

RG 8.20, "Applications of Bioassay for I-125 and I-131"

RG 8.26, "Applications of Bioassay for Fission and Activation Products"

RG 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants"

RG 8.28, "Audible-Alarm Dosimeters"

RG 8.29, "Instructions Concerning Risks From Occupational Radiation Exposure"

The staff is revising some existing RGs and developing additional RGs to address some of the new issues in the revised 10 CFR Part 20. Some of the new or revised RGs that pertain to Chapter 12 of CESSAR-DC are listed below:

RG 8.7, "Instructions for Recordkeeping and Recording Occupational Radiation Exposure Data" (Revised)

RG 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program" (Revised)

RG 8.25, "Air Sampling in the Workplace"

RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses"

RG 8.35, "Planned Special Exposures"

RG 8.36, "Radiation Dose to the Embryo/Fetus"

RG 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants"

Since addressing these RGs is outside the scope of this design certification review, the staff has designated this omission as COL Action Item 12.1.3-1. In any application for a COL, the COL applicant shall state whether it will follow the guidance in these RGs or state what alternative guidance will be used.

ABB-CE's policy and design considerations meet the criteria of SRP Section 12.1 and are acceptable. However, the submittal of detailed operational considerations regarding the implementation of a radiation protection program is outside the scope of the design certification document. COL applicants seeking an operating license by referencing the System 80+ certified design will be

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required to address these operational considerations to the level of detail recommended in RG 1.70 (as discussed herein in Section 12.1.1).

The DSER for Section 12.1 identified a single Open Item 12.1.2-1. ABB-CE addressed this open item, and the staff considers it resolved, as discussed in Section 12.1.2 of this report. The staff's review of Section 12.1 of CESSAR-DC is complete, and no open items remain unresolved.

12.2 Radiation Sources

The staff reviewed the descriptions of the radiation sources given in CESSAR-DC Section 12.2 and CESSAR-DC Chapter 11 for completeness against the guidelines in RG 1.70, and against the criteria in SRP Section 12.2. The contained source terms are used as the basis for the radiation design calculations and for personnel dose assessment. The airborne radioactive source terms are used in the design of ventilation systems and for assessing personnel dose. The staff confirmed that ABB-CE had either committed to follow the guidelines of the RGs and staff positions given in SRP Section 12.2 or had given acceptable alternatives.

12.2.1 Contained Sources

In CESSAR-DC Section 12.2.1, ABB-CE describes plant components that can become significant sources of radiation during plant operations, including shutdown. To calculate the source terms used for shielding design, ABB-CE assumed 0.25-percent fuel cladding defects at full-power operation. The principal source of radiation in the containment, other than the reactor core, is the reactor coolant system. Sources of radiation in the reactor coolant system are fission products released from defective fuel cladding, activation products, and corrosion products. Of these radiation sources, the activation product nitrogen-16 (N-16) is the predominant radionuclide in the reactor coolant pumps, steam generators, and reactor coolant piping during plant operations. The staff reviewed ABB-CE's estimates of N-16 activity levels in various parts of the reactor coolant system and found them comparable to activity levels measured at operating plants. ABB-CE based its estimates of corrosion product activity on the assumptions contained in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," April 1975. After plant shutdown, the predominant long-term sources of radiation in the containment are the spent fuel assemblies.

All large contained sources of radiation in the reactor building subsphere and nuclear annex, and in the fuel,

turbine, auxiliary, and radwaste buildings are listed in CESSAR-DC. For each of these contained sources, the list includes the associated maximum activity levels by isotope. However, at the current stage in the System 80+ design, ABB-CE does not have the specifications for the "as-built" systems and the "as-procured" hardware for a completed plant. Therefore, ABB-CE cannot describe the radioactive system components, which will be significant in-plant radiation sources, to the level of detail specified in RG 1.70 and in SRP Section 12.2. This was designated DSER Open Item 12.2.1-1 (see also Section 12.1.2 of this evaluation). Although the staff needs these details, such as radioactivity content, source geometry, and plant location, to verify the adequacy of the radiation shielding provided, the staff has determined that providing this information goes beyond the design requirements specified in 10 CFR Part 52.

To address this open item, ABB-CE submitted tier-one information design acceptance criteria (DACs) in Table 3.2-1 of ABB-CE's CDM. The purpose of these DACs is to ensure that the COL holder verifies the adequacy of the plant shielding as built before fuel is loaded in the reactor. These DACs specify the methods and assumptions for determining the actual source terms and shielding calculational methods to be used. These source terms and calculational methods will be verified by the inspections, tests, analyses, and acceptance criteria (ITAACs) during plant construction. The methods and assumptions used to calculate the source terms in Section 12.2 and the methods and assumptions specified in the DACs are consistent with the SRP acceptance criteria and will ensure that the System 80+ plant meets the relevant requirements of 10 CFR Part 20, 10 CFR 50.34(f), and General Design Criterion (GDC) 61. DSER Open Item 12.2.1-1 is, therefore, resolved by ABB-CE's issuance of DAC 3.2. The DAC process is evaluated in Section 14.3 of this report.

12.2.2 Airborne Radioactive Material Sources

In CESSAR-DC Section 12.2.2, ABB-CE discusses the sources of airborne radioactivity for the System 80+ design. Airborne radioactive source terms are used in the design of ventilation systems and for personnel dose assessment. In RG 1.70, the staff states that this section should include a tabulation, by nuclides, of the calculated concentrations of airborne radioactive material expected during normal operation and AOOs for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. The level of design detail in the CESSAR-DC does not include system layouts within rooms or cubicles, or detailed information about the type and size of components in these systems. Since leakage characteristics of this unidentified equipment are not known, the concentrations of airborne radioactive material

in equipment rooms or cubicles cannot be given. This lack of information was identified as DSER Open Item 12.2.2-1 (see also Section 12.3.4 of this chapter). To address this open item, ABB-CE has listed DACs in Table 3.2-1 of its CDM. These DACs will require the COL applicant to determine source-term parameters that will be verified by the ITAACs during plant construction. DACs describing the bases for the source term are consistent with the SRP acceptance criteria. Compliance with DACs, supplemented with the information in CESSAR-DC Sections 12.2 and 12.3, will demonstrate that the System 80+ design can meet the related requirements of 10 CFR Part 20, 10 CFR 50.34(f), and GDC 61. DSER Open Item 12.2.2-1 is, therefore, resolved by ABB-CE's issuance of DAC 3.2. The DAC process is evaluated in Section 14.3 of this report.

12.2.3 Sources Used in NUREG-0737 Postaccident Shielding Review

The initial core releases that will be used to determine postaccident radiation levels will be equivalent to the source terms recommended in draft NUREG-1465, RG 1.7, and SRP Section 15.6.5. This is in accordance with 10 CFR 50.34(f)(2)(vii) (Item II.B.2 of NUREG-0660 and NUREG-0737). Item II.B.2 of NUREG-0737 states that applicants should identify systems that contain high levels of radioactivity in postaccident situations. Since ABB-CE does not have the specifications for either the as-built systems or the as-procured hardware for a completed plant, CESSAR-DC Section 12.2.3 does not list such postaccident sources. This lack of information was identified as DSER Open Item 12.2.3-1. To address this open item, ABB-CE has listed DACs in Table 3.2-1 of its CDM. These DACs specify the methods and assumptions for determining the postaccident source terms and state that analyses will be performed to determine the radiation levels in areas which require access for mitigation of or recovery from a design-basis accident. The methods and assumptions used to calculate the postaccident source terms in Section 12.2, and the methods and assumptions specified in the DACs are consistent with the SRP acceptance criteria and will ensure that the System 80+ plant meets the requirements of 10 CFR 50.34(f)(2)(vii) (Item II.B.2 of NUREG-0660 and NUREG-0737). DSER Open Item 12.2.3-1 is, therefore, resolved by ABB-CE's issuance of DAC 3.2. The DAC process is evaluated in Section 14.3 of this report.

In DSER Section 12.2, the staff designated the following as open items: Open Items 12.2.1-1 (also discussed in Section 12.1), 12.2.2-1 (also discussed in Section 12.3), and 12.2.3-1 (also discussed in Section 12.3). All three of these open items were subsequently addressed and resolved by the issuance of DAC 3.2. On this basis, the staff's

review of Section 12.2 of CESSAR-DC is complete and no open items remain unresolved.

12.3 Radiation Protection Design

The staff reviewed the facility design features, shielding, ventilation, and area and airborne radiation monitoring instrumentation described in CESSAR-DC for adherence to the guidelines in RG 1.70 and the criteria in SRP Section 12.3. This review ensured that ABB-CE had either committed to follow the guidelines of the regulatory guides and staff positions referenced in SRP Section 12.3 or given acceptable alternatives. In addition, the staff selectively reviewed the System 80+ design features for radiation protection against the acceptance criteria of the SRP, using the review procedures given there.

12.3.1 Facility Design Features

The facility design features incorporated into the System 80+ design are intended to help maintain occupational radiation exposures ALARA in accordance with the guidance in RG 8.8. The design features are based on the ALARA design considerations described in CESSAR-DC Section 12.1.

The reactor coolant pumps (RCPs) for the System 80+ design use cartridge-type RCP seals that are reliable and easy to replace. Permanent platforms around the RCPs further facilitate seal replacement, thereby reducing maintenance time and exposure. Steam generators in the System 80+ plant are designed to use automatic/robotic equipment for inspection and maintenance activities. In addition, these steam generators have adequate pull and laydown areas, permanent platforms, hand holes, 53-cm (21-in.) manways, and removable insulation to improve accessibility and reduce overall exposure during maintenance and inspection activities. Mechanical snubbers rather than hydraulic snubbers are used in radiation areas to reduce maintenance and inspection needs. Slurry piping systems have remote backflushing capabilities to reduce personnel exposure during servicing. Pumps and connected piping are flanged, where feasible, to facilitate pump removal. Floor drains connecting rooms that have significantly different airborne radioactivity levels are separated or have water-filled loop seals to prevent cross-contamination.

Like the equipment, the layout is designed to reduce personnel exposures. Each inspection and maintenance station has adequate work and laydown space. Rigging and lifting equipment is provided to facilitate the removal, transport, or replacement of equipment or portable shielding during maintenance activities. To improve

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worker efficiency, adequate illumination and support services (e.g., power, service air, water, ventilation, and communications) are available at work stations. Strippable coatings or high-pressure water are used to decontaminate refueling canal surfaces after refuelings. Tube pull areas for components that handle radioactive fluids are designed with curbs, drains, and coated floors to prevent the spread of contamination in the event of spills. Valves associated with highly radioactive components are separated from other components and are located in shielded valve galleries. Radioactive piping is not field routed, but is routed through pipe chases to minimize employee exposures. These equipment and layout design features conform with the guidelines of RG 8.8 for maintaining occupational radiation exposures ALARA and are acceptable.

The System 80+ design incorporates several features to minimize the buildup, transport, and deposition of activated corrosion products in the reactor coolant and auxiliary systems. Except in cases where no proven alternative exists, materials in contact with the primary coolant have low cobalt impurities and low nickel content to reduce the amounts of cobalt-60 and cobalt-58 introduced in the reactor coolant system. Cobalt and nickel levels are reduced or eliminated in bearing journals, valve seats, and steam generator tubes. The System 80+ design minimizes the presence of antimony in the RCP journal bearings. The presence of antimony in RCP journal bearings in some current generation plants has resulted in an increase in the number of hot particles at these plants. Crud traps in welds are minimized by using butt welds in lieu of socket welds. Pump casing drain lines and valves in radioactive service have smooth internal surfaces to minimize internal radioactivity deposition. Tanks containing radioactive liquid have drain pipes connected at the lowest part of the tank and have a convex or sloped-bottom design to minimize radioactivity deposition. Piping systems used to transport process resins are designed to minimize pipe plugging. Equipment and piping containing radioactive materials has provisions for draining and flushing. These methods for reducing crud are based on the guidelines in RG 8.8 and are acceptable.

At the staff's request, ABB-CE furnished oversized drawings of the plant layout that indicate the five radiation zones used in the plant design. These radiation zones serve as a basis for classifying occupancy and access restrictions for various areas within the plant during normal operations and accident conditions. Maximum design dose rates established for each zone are used as input for shielding of the respective zones. This method of plant zoning is consistent with the guidance in RG 1.70 and the SRP and is generally acceptable to the staff. However, since ABB-CE has not given the staff sufficient

information regarding shielding design (see Section 12.3.2), the staff is unable to verify the zone designations given in the CESSAR-DC. This was designated as DSER Open Item 12.2.1-1 (see Section 12.2.1 of this chapter). As stated in Section 12.2.1, ABB-CE addressed this open item in Table 3.2-1 of its CDM.

During the review of the plant layout drawings, the staff identified seven apparent deficiencies, which were categorized as DSER Open Items 12.3.1-1 through 12.3.1-7. These items are discussed below.

When the staff prepared the DSER, the CESSAR-DC did not have detailed plant layout drawings with radiation zone designations for the radwaste, turbine, and service buildings. This deficiency was designated as DSER Open Item 12.3.1-1. In Amendment Q to CESSAR-DC, ABB-CE furnished the requested layout drawings with radiation zone designations. Therefore, DSER Open Item 12.3.1-1 is resolved.

The plant layout drawings in the CESSAR-DC did not initially indicate the major personnel traffic patterns used to access plant areas during normal operations and used to access vital areas during postaccident conditions. This was designated DSER Open Item 12.3.1-2. In subsequent amendments to CESSAR-DC, ABB-CE revised the figures in Chapter 1 to indicate portions of the plant that needed to be accessed for routine plant maintenance. In Amendment Q to CESSAR-DC, ABB-CE modified the plant layout drawings in Chapter 12 to show personnel routes used to access vital areas during postaccident conditions. Therefore, DSER Open Item 12.3.1-2 is resolved.

The initial plant layout drawings did not clearly show the locations of personnel locker and changeout rooms. This was designated DSER Open Item 12.3.1-3. Since the location of these areas was later clarified by ABB-CE, DSER Open Item 12.3.1-3 is resolved.

The personnel decontamination area on elevation 115 + 6 of the reactor building did not contain an uncontaminated exit point to the uncontrolled area. This was designated DSER Open Item 12.3.1-4. In Amendment Q to CESSAR-DC, ABB-CE revised the layout of this area to correct this problem. Therefore, DSER Open Item 12.3.1-4 is resolved.

Several areas in the maintenance/outage area could apparently only be accessed through areas that had higher radiation zones. This was designated DSER Open Item 12.3.1-5. ABB-CE modified the CESSAR-DC in Amendment R to state that the areas in question typically require infrequent access. When access to these areas is required, radiation protection personnel would conduct

radiation surveys of the areas to detect any high radiation areas. Temporary shielding, or other controls, will then be used, if necessary, to minimize doses to personnel requiring access to these areas. Therefore, DSER Open Item 12.3.1-5 is resolved.

The access area to the spent fuel transfer tube on the 115 + 6 elevation of the reactor building has the potential for lethal dose rates when a spent fuel assembly is transferred through the adjacent portion of the spent fuel transfer tube. The initial plant layout drawing depicting this access area was not sufficiently detailed for the staff to judge the adequacy of the access area shielding and surrounding dose rates. This lack of detail was designated DSER Open Item 12.3.1-6. ABB-CE subsequently gave the staff detailed layout drawings depicting the cubicle dimensions and shielding used in this access area. These drawings show the addition of a wire cage with locked door to restrict access to the spent fuel transfer tube. These revised layout drawings, however, do not adequately depict the radiation levels in the access area to the spent fuel transfer tube. To address this issue, ABB-CE has submitted DACs in Table 3.2-1 of the CDM. The purpose of these DACs is to ensure that the adequacy of the plant shielding, as built, is verified by the COL holder before fuel is loaded in the reactor. Compliance with these DACs, supplemented with the information in CESSAR-DC Sections 12.2 and 12.3, will demonstrate that the System 80+ design can meet the related requirements of 10 CFR Part 20, 10 CFR 50.34(f), and GDC 61 which pertain to the adequacy of the plant shielding and fuel storage, both during normal operating and postaccident conditions. Therefore, DSER Open Item 12.3.1-6 is resolved.

At the time of the DSER review, the plant layout drawings depicted the primary chemistry lab area on elevation 50 as a radiation zone 3 area. Since this lab will be frequently occupied during normal operations and is classified as a vital area, it should be designated as a radiation zone 2 area. This was designated DSER Open Item 12.3.1-7. In Amendment Q to CESSAR-DC, ABB-CE revised the radiation zone designation for this laboratory area from a zone 3 to a zone 2 area. This revision effectively resolved DSER Open Item 12.3.1-7.

Source-term information that should be used for calculating postaccident radiation levels is described in 10 CFR 50.34(f)(2)(vii) (Item II.B.2 of NUREG-0660 and NUREG-0737). This section also states that the postaccident plant dose rates should be such that the dose to plant personnel should not exceed 5×10^{-2} sieverts (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident (per GDC 19). The dose rate in areas requiring continuous occupancy (vital areas) should be less than 15×10^{-5} sieverts/hr (15 mrem/hour) averaged over 30 days. In the CESSAR-DC, ABB-CE

states that personnel exposures will meet GDC 19 and NUREG-0737 guidelines. Issue II.B.2 of NUREG-0737 also recommends that the CESSAR-DC list all vital areas (areas requiring continuous occupancy during the course of an accident) in the plant. In CESSAR-DC, ABB-CE initially gave only a partial listing of System 80+ vital areas. This, along with other missing information described in Section 12.2.3 of this evaluation, was designated DSER Open Item 12.2.3-1. In Amendment S to CESSAR-DC, ABB-CE gave more complete listing of vital areas, resolving the staff's concern. A complete list of all plant vital areas will be developed by the COL applicant on the basis of site-specific emergency procedures (COL Action Item 12.3.1-1). The staff's other concerns in DSER Open Item 12.2.3-1 were addressed in Table 3.2-1 of ABB-CE's CDM.

In Item II.B.2 of NUREG-0737, the staff states that an applicant must summarize the integrated doses to personnel in each of the plant areas requiring either continuous occupancy or infrequent access for the duration of the accident (these doses should include exposure received while in transit between vital areas) and a listing of the dose rates in these areas 1 hour, 1 day, 1 week, and 1 month after an accident. ABB-CE did not give this information in CESSAR-DC. This lack of was designated DSER Open Item 12.2.3-1 (other parts of this open item are addressed in Sections 12.2.3 and 12.3.2 of this evaluation). In response to this open item, ABB-CE submitted DACs in Table 3.2-1 of its CDM. These DACs specify the methods and assumptions for determining the postaccident source terms and state that the COL applicant will perform analysis to determine the radiation levels in areas that require access for mitigation of or recovery from a design-basis accident. On the basis of the information in DAC Table 3.2-1, DSER Open Item 12.2.3-1 is resolved. The DAC process is evaluated in Section 14.3 of this report.

12.3.2 Shielding

The function of the plant's radiation shielding is to protect plant personnel and the public against radiation exposure from the various sources of ionizing radiation in the plant during normal operation (including AOOs and maintenance) and during accident conditions. The System 80+ design also includes shielding, where required, to mitigate the possibility of radiation damage to materials. Radioactive components and piping will be separated from non-radioactive components and piping to minimize personnel exposure during maintenance and inspection activities. Major radioactive piping will be located in shielded pipe chases. Where necessary pumps and other support equipment for components that contain radioactive

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material will be located outside the component cubicle in separate shielded cubicles. Shielded compartments will have labyrinth entrances to minimize radiation streaming directly through access openings. Cubicles containing radioactive materials will be shielded overhead to minimize skyshine. Penetrations will be located so that the radioactive source will not be in a direct line to adjacent areas that may have employees in them. Space will be allocated, where needed, for the erection of temporary shielding. These shielding techniques comply with the shielding guidelines contained in RG 8.8 and are acceptable.

ABB-CE states that the certification document will not include a description of the physical dimensions and compositions of the radiation shielding in the System 80+ design since this information will not be available at the time of design certification. Since RG 1.70 and the acceptance criteria of the SRP require this information be provided to permit the staff to conduct confirmatory calculations of shielding effectiveness, the staff designated this as part of DSER Open Item 12.2.3-1 (see Sections 12.2.3 and 12.3.1 of the evaluation for other parts of this open item). As stated earlier, ABB-CE submitted DACs in Table 3.2-1 of its CDM to respond to this open item. These DACs require the COL applicant to verify the adequacy of (1) the shielding around rooms, corridors, and operating areas during normal operations and shutdown conditions, (2) the shielding in plant areas where maintenance is performed, and (3) the shielding around areas that must be accessible for mitigation of or recovery from a design-basis accident. The analysis assumptions, methods, and acceptance criteria in these DACs are consistent with the criteria in the SRP. Therefore, the staff concludes that compliance with these DACs, as supplemented with the information contained in CESSAR-DC Section 12.3.2, is acceptable to adequately address the relevant requirements of 10 CFR Part 20, 10 CFR 50.34(f)(2)(vii) (Item II.B.2 of NUREG-0660, and NUREG-0737), and GDC 61 which require that shielding design reviews be performed to ensure adequate access to necessary plant areas during normal operating and postaccident conditions.

There have been several overexposures or near overexposures at pressurized-water reactors (PWRs) in recent years. Several overexposures have occurred near the spent fuel transfer tube. People working in areas adjacent to the spent fuel transfer tube can be exposed to potentially lethal levels of radiation when spent fuel assemblies pass through this tube. Overexposures have also occurred in the reactor cavity area. In CESSAR-DC, ABB-CE did not list accessible plant areas where, during normal operations and AOOs, personnel could receive 1 gray (100 rad) or more in 1 hour; this was designated DSER Open Item 12.3.2-1. In response to this open item,

ABB-CE, as part of Amendment S to the CESSAR-DC, added a table listing all accessible plant areas where personnel could receive 1 gray (100 rad) or more in 1 hour. ABB-CE revised the CESSAR-DC, in Amendment Q, to include a detailed layout drawing of the access hatch to the spent fuel transfer tube. ABB-CE also described the configuration of the shielding around the spent fuel transfer tube access hatch to minimize doses to personnel from spent fuel bundles being transported through the tube. Another accessible plant area that could have dose rates of 1 gray (100 rad) or more in 1 hour is the reactor cavity incore chase during incore instrument withdrawal. In Amendment T to CESSAR-DC, ABB-CE described the positive access controls used to prevent personnel entry to this area during instrument withdrawal. Access to the incore chase is controlled by a lockable access door. This door has a warning light that lights up when the incore instrumentation is being withdrawn. In addition, an electrical interlock will prevent personnel entry to the incore chase through this door during instrument withdrawal. When the incore instrumentation are withdrawn, radiation protection personnel will post a high-radiation sign outside the entrance to the incore chase area. An area radiation monitor located in the incore chase gives indication and alarm during withdrawal of incore instrumentation. On the basis of this additional information, DSER Open Item 12.3.2-1 is resolved.

The SRP states that an applicant must describe how the shielding parameters are determined, including pertinent codes, assumptions, and techniques used in the shielding calculations. In response to an August 3, 1991, staff request, ABB-CE revised the CESSAR-DC in Amendment J to describe the shielding codes to be used to determine the adequacy of the station shielding design. ABB-CE will use the shielding codes ANISN, DOT, MORSE, and SABINE to verify the effectiveness of the primary shield (around the reactor core). In CESSAR-DC, ABB-CE also describes shielding codes used to verify gamma-source shielding elsewhere in the plant. The ANISN, DOT, and MORSE calculational codes are recommended in the SRP and are acceptable to the staff. SABINE is an updated code that is also acceptable to the staff. The SABINE code is used to perform neutron and gamma ray shielding analyses with removal-diffusion methods. This is a commonly acceptable shielding code that has been used and accepted by NRC to design the shielding in several operating nuclear plants in the U.S.

12.3.3 Ventilation

The System 80+ ventilation systems are designed to protect people and equipment from extreme environmental conditions and to ensure that exposure to airborne

radioactivity levels is minimized and maintained ALARA. Design features incorporated to maintain personnel exposures ALARA include the following:

- (1) Ventilation air is supplied directly to the clean areas of the plant and the air from the potentially contaminated areas is exhausted, thereby creating a positive flow of air from the clean areas to the potentially contaminated areas.
- (2) Appropriate use of negative and positive pressure in plant areas respectively prevents exfiltration or infiltration of possible airborne radioactive contamination.
- (3) A dual fresh-air-filtered intake system for the control room ventilation supplies uncontaminated air regardless of wind direction.

These design criteria conform to the guidelines of RG 8.8 and are acceptable to the staff. However, ABB-CE does not list the concentrations of airborne radioactive contamination in cubicles, rooms, and corridors. This was designated DSER Open Item 12.2.2-1. However, as discussed in Section 12.2.2 of this chapter, this open item was resolved through the issuance of the DACs in Table 3.2-1 of ABB-CE's CDM. These DACs require the COL applicant to perform analyses to predict airborne concentrations in rooms, corridors, equipment cubicles, and operating areas requiring personnel access during normal operations and resolve DSER Open Item 12.2.2-1.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The area radiation monitoring equipment alerts operators and other station personnel of changing or abnormally high radiation conditions in the plant to prevent possible overexposures. Control room displays give information on monitor readings, alarm setpoints, and operating status. The area radiation monitors are comprised of microprocessors and Geiger-Mueller tubes or ionization chambers for detecting gamma-radiation. Some plant areas may require extended or high-range detector configurations to cover special operational or postaccident monitoring functions. Area radiation monitors will have both local visual and audible alarms. Area radiation monitors located in high noise areas may have additional visual indication, if necessary, to ensure that nearby personnel promptly recognize high-radiation conditions. Area radiation monitors are calibrated once every refueling cycle. This description of the System 80+ area radiation monitoring system is acceptable since it meets the acceptance criteria in SRP Section 12.3 regarding area and airborne radiation

monitoring description, location, sensitivity, readout and calibration.

ABB-CE states that the area radiation monitors are located according to the potential for significant radiation levels in an area and according to the expected occupancy of the area. Area radiation monitors will also be located in places where postaccident access to safety-related equipment may be required and around new fuel handling and storage areas for criticality accident detection to meet the requirements of 10 CFR 70.24. Area radiation monitors are not located in high-radiation areas that are normally not accessible or in frequently accessed areas that are typically not radioactive. In SRP Section 12.3, the staff states that an applicant should locate fixed area radiation monitors in accordance with American National Standards Institute (ANSI)/American Nuclear Society-HPSSC-6.8.1. Because ABB-CE had not listed where fixed area radiation monitors were located when the DSER was prepared, this was designated DSER Open Item 12.3.4-1. In response to this open item, ABB-CE has given, in Amendments Q and S, a list of generic locations of the fixed area radiation monitors that will be used in the System 80+ plant design. ABB-CE states that the exact area radiation monitor locations cannot be determined until information on pipe routing, equipment location, and equipment leak rates is known. This generic listing of fixed area radiation monitor locations is acceptable and DSER Open Item 12.3.4-1 is resolved. COL applicants seeking an operating license by referencing the System 80+ certified design will be required to give the exact locations of these radiation monitors (COL Action Item 12.3.4-1).

In Section II.F.1-3 of NUREG-0737, the staff recommends that the reactor containment be equipped with two physically separated radiation monitoring systems that are capable of measuring up to 0.72 c/kg/sec (10^7 R/hr) in the containment after an accident. In the CESSAR-DC, ABB-CE states that the System 80+ design incorporates two physically separated and electrically independent ion chambers located inside the reactor containment to measure high-range gamma radiation. The design and qualification of these monitors complies with the guidelines of RG 1.97. ABB-CE states that the high-range containment area monitors also meet all of the recommendations of 10 CFR 50.34(f)(2)(xvii) (Item II.F.1-3 of NUREG-0660 and NUREG-0737), including detector range, response, redundancy, separation, onsite calibration, and environmental design qualification. However, since ABB-CE had not specified the location of these high-range monitors on the plant layout drawings at the time of the DSER review, the lack of monitor location was designated DSER Open Item 12.3.4-2. In Amendments Q and S, ABB-CE has since specified the location of the contain-

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ment high-range monitors on the plant layout drawings. Therefore, DSER Open Item 12.3.4-2 is resolved.

The airborne radiation monitoring equipment is in selected areas and ventilation systems to give plant operating personnel continuous information about the airborne radioactivity levels throughout the plant. ABB-CE states that the airborne radioactivity monitors are upstream of the filter trains so that they monitor representative radioactivity concentrations from the areas being sampled. However, since the airflow diagrams in CESSAR-DC did not initially indicate the locations of the airborne radioactivity monitors, this was designated DSER Open Item 12.3.4-3. In Amendment Q, ABB-CE has modified the ventilation flowpath figures in CESSAR-DC to indicate the locations of these airborne radioactivity monitors. On this basis, DSER Open Item 12.3.4-3 is resolved.

In response to an August 3, 1991, staff request, ABB-CE states that plant airborne radioactivity monitors are capable of detecting the time-integrated change of the most limiting particulate and iodine species equivalent to 10 CFR Part 20, Appendix B concentrations in each monitored plant area within 10 MPC-hours. This monitor sensitivity is in agreement with the monitor sensitivity criteria in SRP Section 12.3. SRP Section 12.3 also states that the CESSAR-DC must provide the criteria and methods for obtaining representative in-plant airborne radioactivity concentrations in all work areas. ABB-CE states that insufficient design information is available to discuss the dilution of air from specific areas or rooms. Consequently, the CESSAR-DC does not list representative airborne radioactivity concentrations for rooms or other areas that may be occupied by employees. This was designated part of DSER Open Item 12.2.2-1. As discussed in Section 12.2.2 of this chapter, this open item was resolved through the issuance of the DACs in Table 3.2-1 of ABB-CE's CDM.

At the time of the DSER review, CESSAR-DC did not describe how the airborne radioactivity monitoring system satisfied the criteria contained in ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities" (as referenced in NUREG-0800). This was identified as DSER Open Item 12.3.4-4. ABB-CE has since amended the CESSAR-DC to state that the airborne radioactivity monitoring system complies with all the associated acceptance criteria contained in ANSI N13.1-1969. Therefore, DSER Open Item 12.3.4-4 is resolved.

In 10 CFR 50.34(f)(2)(xxvii) (Issue III.D.3.3 of NUREG-0660 and NUREG-0737), each applicant is required to provide equipment and associated training and procedures for accurately determining the airborne iodine concentrations in areas within the facility where personnel

may be present during an accident. In response to an August 3, 1991, staff inquiry, ABB-CE stated that a portable airborne monitor will be available to allow accurate determination of airborne iodine concentrations in potentially occupied areas that would not be directly covered by fixed instrumentation. ABB-CE stated that this portable airborne monitor will also meet the equipment recommendations given in Issue III.D.3.3 of NUREG-0737, including recommendations on sample media, purging, and calibration. This commitment complies with the criteria in Issue III.D.3.3 of NUREG-0737. However, COL applicants seeking an operating license by referencing CESSAR-DC should provide additional information concerning specific equipment to be used and procedures that will be followed to implement Issue III.D.3.3 of NUREG-0737. This is COL Action Item 12.3.4-2.

In DSER Section 12.3, the staff designated the following 12 open items; 12.3.1-1, 12.3.1-2, 12.3.1-3, 12.3.1-4, 12.3.1-5, 12.3.1-6, 12.3.1-7, 12.3.2-1, 12.3.4-1, 12.3.4-2, 12.3.4-3, and 12.3.4-4. On the basis of subsequent information submitted by ABB-CE, the staff considers all of these resolved, as discussed in the foregoing sections. DSER Section 12.3 also contained portions of DSER Open Items 12.2.2-1 and 12.2.3-1. As discussed in the staff's evaluation, these two open items were addressed and resolved by the issuance of DAC 3.2.

12.4 Dose Assessment

The staff has reviewed ABB-CE's dose assessment for the System 80+ design for completeness against the guidelines in RG 1.70 and against the criteria in SRP Section 12.4. This review consisted of ensuring that ABB-CE has either committed to follow the criteria of the RGs and staff positions in SRP Section 12.4 or has given acceptable alternatives. In addition, the staff selectively compared ABB-CE's dose assessment for specific functions against the experience of operating PWRs.

In CESSAR-DC Section 12.4, ABB-CE describes design features that will be incorporated into the System 80+ design to ensure that occupational radiation exposures are maintained ALARA and no more than the goal of 1 person-sievert/yr (100 person-rem/yr). Many of these ALARA design features were described earlier in CESSAR-DC Chapter 12. In order to reduce the source of cobalt in the primary system, cobalt alloys and cobalt-based hardfacing materials are minimized (except in cases where no proven alternative material exists) in all primary system materials in contact with primary coolant. Piping and equipment in direct contact with the reactor coolant system have a cobalt content of less than 0.05 wt percent. Steam generator tubes have a cobalt content of less than

0.015 wt percent. The System 80+ design minimizes the use of antimony in the RCP journal bearings. In addition, all other pump parts that are wetted by reactor coolant have minimal or no antimony content. Leakage from the System 80+ fuel is expected to be less than 0.1 percent. The System 80+ design specifies the use of elevated pH levels in the primary system. Elevated pH levels reduce equilibrium corrosion rates and the buildup of activated corrosion products on primary system surfaces. Such features reduce the source term and, therefore, the radiation levels in the primary system. Other System 80+ design features described in this section of CESSAR-DC include the use of an extended fuel cycle, improved equipment accessibility, and the use of more reliable equipment to reduce the frequency of maintenance work. Such features, in use at many currently operating power plants, are intended to reduce occupational radiation exposures and are acceptable.

ABB-CE estimates that the average annual collective dose for the System 80+ design will be 7.9×10^{-1} person-sieverts (79 person-rem). In arriving at this estimate, ABB-CE used 1989 dose data, broken down by work group and task, for seven PWRs operated by Duke Power. The resulting dose estimates for each work group and task category were then multiplied by dose reduction factors to obtain dose estimates for the System 80+ design. ABB-CE states that these dose reduction factors, which were based on engineering judgment, took into account the numerous ALARA design features incorporated into the System 80+ design. Because ABB-CE did not adequately explain the basis for determining these ALARA dose reduction factors, this was designated DSER Open Item 12.4-1. In Amendments J and S to the CESSAR-DC, ABB-CE modified these dose reduction factors and justified their use in a more detail. On the basis of this new information, DSER Open Item 12.4-1 is resolved.

The level of detail of ABB-CE's dose assessment is inconsistent with the level of detail prescribed in the acceptance criteria of Section 12.4 of the SRP, nor does it meet the intent of RG 8.19. Both of these documents require detailed information as to expected occupancy of plant radiation areas as well as estimated doses associated with major job and work functions. In response to the staff's concerns, ABB-CE stated that a formal dose assessment could not be made until a detailed shielding analysis was performed, in-plant radioactive airborne concentrations were calculated, and information was obtained on plant occupancy times and reliability data. ABB-CE's DACs in Table 3.2-1 of the CDM require that the COL applicant perform a detailed shielding analysis and airborne concentration calculation before fuel is loaded into the reactor. The DAC process is evaluated in Section 14.3 of this report. Compliance with these DACs will

ensure that the COL applicant will have sufficient information to perform a formal dose assessment. This is COL Action Item 12.4.5-1.

In Section 12.4 of RG 1.70, the staff states that, for areas with expected airborne radioactivity concentrations, an applicant list the estimated inhalation exposure to personnel. Since this information was not included in Section 12.4 of CESSAR-DC, this was designated DSER Open Item 12.4-2. ABB-CE states that this information will not be given as part of the design certification since the information necessary to calculate airborne radioactivity levels is not available at this stage of the design. However, ABB-CE's DACs in Table 3.2-1 of the CDM require that the COL applicant perform an analysis to predict airborne radioactive concentrations in rooms, corridors, equipment cubicles, and operating areas throughout the plant before fuel is loaded. Compliance with these DACs will ensure that the COL applicant will have sufficient information to calculate estimated inhalation exposures to plant personnel. On the basis of these DACs, DSER Open Item 12.4-2 is resolved.

The DSER for CESSAR Section 12.4 designated Open Items 12.4-1 and 12.4-2. On the basis of subsequent amendments to the CESSAR-DC and the submittal DAC 3.2, both of these open items are resolved.

12.5 Organization

The organization required to implement an effective health physics program and ensure that radiation exposures are within the limits of 10 CFR Part 20 and are ALARA is outside the scope of this review. COL applicants seeking an operating license by referencing the System 80+ certified design will be required to address this concern to the level of detail discussed in RG 1.70. This is COL Action Item 12.5-1.

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13.1 Organizational Structure of Site Operator

This section describes the staff's final review and evaluation of ABB-CE's CESSAR-DC Section 13.1, "Organizational Structure of Site Operator," and finds that ABB-CE's statement related to the combined license (COL) organizational structure is acceptable. The staff agrees that the information in this area is outside the scope of the System 80+ standard plant design. The COL applicant referencing the System 80+ certified design will be required to submit site-specific information at the COL phase described in 10 CFR 52.79(b). COL Action Item 13.1-1 is acceptable.

Shift Staffing

By letter dated December 18, 1992 (LD-92-120), ABB-CE addressed Generic Safety Issue (GSI) HF1.1, "Shift Staffing." ABB-CE indicated that the System 80+ Technical Specifications (CESSAR-DC Chapter 16) identify minimum shift staffing requirements (Section 5) in accordance with NUREG-1432, "Standard Technical Specifications for Combustion Engineering Plants" (January 1991). Further, ABB-CE noted that the COL applicant will document implementation of this requirement in accordance with Regulatory Guide (RG) 1.114 as part of the site organization. The staff interprets this requirement to be a part of COL Action Item 13.1-1. On this basis, COL Action Item for GSI HF1.1 is acceptable.

13.2 Training

This section describes the staff's final review and evaluation of CESSAR-DC Section 13.2, "Training." The review was based on current regulatory requirements established in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f), and the guidance contained in Standard Review Plan (SRP) Sections 13, NUREG-0700, "Guidelines for Room Design Reviews" (September 1981), and NUREG-0933, "A Prioritization of Generic Safety Issues" (August 1987). Additional review criteria were developed to provide a basis for review of aspects of the ABB-CE System 80+ human factors engineering (HFE) program that were not fully addressed by the previously mentioned documents. These criteria are contained in the staff's "HFE Program Review Model and Acceptance Criteria," (HFE PRM), which was forwarded to the Commission in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design" (August 27, 1992). The HFE PRM considered aspects of training as it pertained to

the verification and validation (V&V) of the main control room (MCR) and remote shutdown system designs.

ABB-CE does not include training development within the scope of the System 80+ design certification application and states that development of training would be the responsibility of the COL applicant. However, in response to the staff's draft safety evaluation report (DSER), information regarding the incorporation of vendor technical information into the plant designer's training information plan as part of the Operational Support Information Program (OSIP) were submitted to the staff for review. The staff reports on its review of OSIP in Chapter 18 of this report.

Although training is not considered to be part of the information required for certification of the System 80+ design, the staff noted several issues related to training during the review of CESSAR-DC Chapter 18, "Human Factors Engineering." Items regarding training materials, the use of a simulator for training, and the incorporation of operational experience into training programs were identified. Each of these items is discussed below.

13.2.1 Training Materials

In the DSER, the staff stated that it "expects CE to develop and submit for certification a detailed program description for developing the training material as part of the design certification for the System 80+." After issuing the DSER, the staff determined that training materials were already a part of the established licensing review process under 10 CFR Part 50 for the COL applicant and, therefore, did not need to be addressed in the System 80+ design certification review under 10 CFR Part 52. The staff agrees that the submittal of training materials is outside the scope of the System 80+ design certification. Therefore, this part of COL Action Item 13.2-1 is acceptable.

13.2.2 TMI Action Item I.A.4.2

10 CFR 50.34(f)(2)(i) corresponds to TMI Action Item I.A.4.2, "Long Term Simulator Training Upgrade," on simulator capabilities. The staff agrees that the information in this area is outside the scope of the System 80+ standard plant design. The COL applicant referencing the System 80+ certified design will be required to submit site-specific information at the COL phase described in 10 CFR 52.79(b). On this basis, this part of COL Action Item 13.2-1 is acceptable.

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13.2.3 TMI Action Item I.C.5

The staff reviewed TMI Action Item I.C.5, "Feedback of Operating Experience," on the incorporation of operational experience into the training and procedure development programs. The staff has determined that development of detailed procedures and training materials is outside the scope of the System 80+ design certification and is the responsibility of the COL applicant. The COL applicant referencing the System 80+ certified design will be required to submit site-specific information at the COL phase described in 10 CFR 52.79(b). ABB-CE has included the training and procedure development process as COL license information items in CESSAR-DC Sections 13.2 and 13.5, respectively. Therefore, part of COL Action Items 13.2-1 and 13.5-1 are acceptable.

13.2.4 TMI Action Item I.G.1

The staff has reviewed TMI Action Item I.G.1, "Training Requirements for Preoperational and Low-Power Testing." The staff determined that I.G.1 is outside the scope of System 80+ certification and is the responsibility of the COL applicant. The COL applicant referencing the System 80+ certified design will be required to submit site-specific information at the COL phase as described in 10 CFR 52.79(b). Therefore, this part of COL Action Item 13.2-1 is acceptable.

In summary, the staff finds ABB-CE's approach to training development as stated in CESSAR-DC Section 13.2 acceptable since it will be addressed by the COL applicant and is identified as a COL responsibility in the System 80+ design certification application. On this basis, the COL Action Item 13.2-1 discussed above is acceptable.

13.3 Emergency Planning

The staff reviewed CESSAR-DC Section 13.3 and ABB-CE's responses to the staff's request for additional information (RAI). By letters dated March 15, 1991, and January 24, January 29, and February 28, 1992, ABB-CE responded. The staff confirmed that the responses to the staff's RAIs were incorporated in Amendment J of CESSAR-DC. This was designated as Confirmatory Item 13.3-1. On the basis of the staff's review of this revision, DSER Confirmatory Item 13.3-1 is acceptable.

In CESSAR-DC Section 13.3, ABB-CE indicates that emergency planning is outside the scope of the System 80+ standard design, and that emergency planning information will appear in the site-specific safety analysis report. The staff agrees that emergency planning will be

addressed by the COL applicant referencing the System 80+ standard design and will significantly depend upon plant-specific and site-specific characteristics. This was designated as COL Action Item 13.3-1 in the DSER.

Nevertheless, certain design features, facilities, functions, and equipment necessary for emergency planning must be considered in the design basis of a standard plant. These are addressed in CESSAR-DC Section 13.3, which describes and gives specifications for the Technical Support Center (TSC) and the Operations Support Center (OSC). The TSC and OSC are shown in Drawing No. 4248-00-1607.00, Revision 1, G200-05, "Nuclear Island General Arrangement Plan at El. 130+6." In Amendment Q to the CESSAR-DC, this drawing appears as Figure 1.2-8.

Section 13.3.3, "BOP Interfaces," of the CESSAR-DC, supplies design considerations for the TSC and OSC which are acceptable in that the facilities and equipment therein are compatible with the control room and meet the applicable criteria of NUREG-0696, "Functional Criteria for Emergency Response Facilities Final Report" (February 1981), which provide the NRC staff positions for acceptable emergency response facility features.

ABB-CE initially considered other facilities that support emergency planning outside the scope of the System 80+ standard design. These included: (1) a near-site Emergency Operations Facility (EOF) for the management of overall licensee emergency response, including coordination with Federal, State, and local officials; (2) onsite decontamination facilities (ODFs) located in the nuclear annex (El. 91+9) and in the radwaste facility; and (3) laboratory facilities to support efforts to monitor and analyze plant systems and environmental samples. In CESSAR-DC Section 13.3, ABB-CE describes a typical EOF, as well as decontamination and laboratory facilities and the interface requirements that must be met to ensure adequacy with the System 80+ standard design. The staff agrees that the EOF is outside the scope of the System 80+ standard design, but must be addressed by a COL applicant referencing the System 80+ standard design. This was designated as COL Action Item 13.3-2. In the DSER, the staff stated that the ODFs and laboratory facilities are within the scope of the System 80+ design. This was designated DSER Open Item 13.3-1. In Amendment J to the CESSAR-DC, additional information was submitted concerning the description and specifications of onsite decontamination and laboratory facilities. In Amendment T to the CESSAR-DC, Table 1.9-1 was amended to reflect layout and equipment for the onsite decontamination and laboratory facilities. On the basis of the staff's review of this revision, COL Action Item 13.3-2 is acceptable.

Additionally, CESSAR-DC Section 1.8 indicates that the Nuclear Regulatory Commission's (NRC's) RG 1.101, "Emergency Planning for Nuclear Power Plants," has been "withdrawn" and is not applicable to the System 80+ design. ABB-CE has stated that the criteria in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (November 1980)/FEMA-REP-1, as endorsed by RG 1.101, serve as the basis for NRC licensees and State and local governments to develop acceptable radiological emergency plans and improve emergency preparedness and are, therefore, not applicable to the System 80+ design. The staff did not agree. "Planning Standards and Evaluation Criteria," Section II.H of NUREG-0654, "Emergency Facilities and Equipment," are applicable to the System 80+ design, as discussed above. In the draft safety evaluation report, the staff stated that ABB-CE should include NUREG-0654 in CESSAR-DC Section 1.8 and CESSAR-DC Table 1.8-1. This was designated as DSER Open Item 13.3-2. In Amendment T to CESSAR-DC, Table 1.8.1 of Section 1.8 was revised to include RG 1.101 and to show NUREG-0654 as applicable with respect to System 80+ emergency response facilities described in Sections 13.3.3.1 and 13.3.3.3 of the CESSAR-DC. On the basis of the staff's review of this revision, DSER Open Item 13.3-2 is resolved.

Design Certification Material

The requirements for the TSC and OSC are covered in the inspections, tests, analyses, and acceptance criteria (ITAACs). This was designated as a generic open item in DSER Open Item 14.1-1, "Review of ITAAC[s] will be provided in the FSER". ABB-CE submitted a revised set of design descriptions and ITAACs. The adequacy and acceptability of the System 80+ design description and ITAACs are discussed in Section 14.3 of this report. On the basis of this evaluation, the emergency preparedness portion of DSER Open Item 14.1-1 is resolved.

10 CFR 50.34.f: TMI-Related Items

CESSAR-DC Section 13.3 addresses TMI Action Plan (NUREG-0660, "NRC Action Plan Development as a Result of TMI-2 Accident," July 1980) Item III.A.1.2 (10 CFR 50.34(f)(xxv)). This item requires licensees to upgrade their emergency support facilities by establishing a TSC, an OSC, and a nearsite EOF for command and control, support, and coordination of onsite and offsite functions during reactor accident situations. As discussed in Section 13.3 of this chapter, the System 80+ design has a TSC and an OSC. The staff considers the nearsite EOF outside the scope of the System 80+ standard plant design. The COL applicant referencing the System 80+ standard

plant design will have to address the nearsite EOF; this has been designated COL Action Item 13.3-2.

13.4 Operational Review

This area is outside the scope of the System 80+ design certification and CESSAR-DC. This area will be addressed by a COL applicant. This is COL Action Item 13.4-1.

13.5 Plant Procedures

This section describes the staff's final review and evaluation of CESSAR-DC Section 13.5. The review was based on current regulatory requirements established in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f), and on the guidance contained in SRP Chapter 13, NUREG-0700 and NUREG-0933. Additional review criteria were developed to provide a basis for review of aspects of the System 80+ HFE program that were not fully addressed by the previously mentioned documents. These criteria are contained in the staff's "HFE Program Review Model and Acceptance Criteria" (HFE PRM), which was forwarded to the Commission in SECY-92-299 dated August 27, 1992. The HFE PRM considered aspects of plant procedures as it pertained to the V&V of the MCR and remote shutdown system designs.

ABB-CE has not included procedure development within the scope of its System 80+ design certification application. Procedure development is identified as a COL responsibility in CESSAR-DC Section 13.5.1. ABB-CE's description of this COL license information item is consistent with the staff's HFE PRM - Element 7, "Procedures", developed to support the review of the System 80+ HFE effort. Details of the staff's review of ABB-CE's application of Element 7 appear in Section 18.7 of this report. This part of COL Action Item 13.5-1 is resolved.

Although plant procedures are not considered to be part of the information required for certification of the System 80+ design, the staff found several issues that were related to procedure development when it reviewed CESSAR-DC Chapter 18, "Human Factors Engineering." Items regarding procedure development, as well as differences between Element 7 of the HFE PRM and CESSAR-DC Section 13.5.1, are discussed below.

Elimination of General Criterion 6

General Criterion 6 of Element 7 of the HFE PRM states: "An analysis shall be conducted to determine the impact of providing computer-based procedures and to specify where such an approach would improve procedure utilization and

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reduce operating crew errors related to procedure use." ABB-CE indicates that an analysis of computer-based procedures is not envisioned as part of the certified design process and that there was no intention to develop computer-based procedures as part of the System 80+ operating ensemble. Because ABB-CE has decided to preclude the development of computer-based procedures as part of the System 80+ design process, the staff considers elimination of this criterion acceptable.

Although plant procedures are not considered to be part of the information required for design certification of the System 80+, the staff found several additional issues related to procedures during the review of ABB-CE's HFE program. Issues regarding procedure guidelines for updating procedures (HF-4.4), short-term accident procedures review (TMI Item I.C.1), incorporation of operating experience (TMI Item I.C.5) and a long-term plan for upgrading procedures (TMI Item I.C.9) were noted. Each of these items is addressed below.

USI/GSI Item HF-4.4, and TMI Action Items I.C.1, I.C.5 and I.C.9

The staff has reviewed HF-4.4, "Guidelines for Upgrading Other Procedures"; TMI Item I.C.1, "Short-term accident and procedures review"; TMI Item I.C.5, "Feedback of Operating Experience"; and TMI Item I.C.9, "Long Term Plan for Upgrading Procedures." The staff determined that development of detailed procedures is beyond the scope of the System 80+ certification and is the responsibility of the COL applicant. ABB-CE has included the procedure development process as a COL license information item in CESSAR-DC Section 13.5.1. CESSAR-DC Section 13.5.1 states that the methods and criteria for the development, V&V, implementation, maintenance, and revision of procedures will include considerations of I.C.1, I.C.5, and I.C.9. On this basis, COL Action Items 13.5-1 is acceptable.

In summary, the staff finds ABB-CE's approach to procedure development as described in CESSAR-DC Section 13.5.1 acceptable since it will be addressed by the COL applicant referencing the System 80+ certified design. The COL applicant will be required to submit site-specific information described in 10 CFR 52.79(b) at the COL stage. COL Action Item 13.5-1 is found to be acceptable.

13.6 Industrial Security

In CESSAR-DC Section 13.6, ABB-CE states that site security is within the scope of the COL applicant. By letter dated February 28, 1992, ABB-CE proposed a new

CESSAR-DC Section 13.6.1 that will specify site-specific interface requirements to be addressed in COL applications that reference the System 80+ design. (These requirements will not be interface requirements in the sense of 10 CFR 52.47(a)(vii), but will be instead action items that must be addressed in the COL applications.)

In CESSAR-DC Section 13.6.1(A), ABB-CE requires the physical security system design to conform to the eight regulatory documents listed in Part II of SRP Section 13.6, Revision 2 (NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981); however, one item did not reflect a recent regulatory change. For employee screening requirements, ABB-CE should reference 10 CFR 73.56 and RG 5.66, "Access Authorization Program for Nuclear Power Plants" (June 1991), in lieu of ANSI N18.17. Updating access authorization references in CESSAR-DC Section 13.6.1(A) was designated as Open Item 13.6-1. By CESSAR-DC Amendment Q, ABB-CE changed the reference for authorization accession in CESSAR-DC Section 13.6.1(A) to 10 CFR 73.56 and RG 5.66. These are the appropriate references and DSER Open Item 13.6-1 is resolved.

In Section 13.6.1(B), a COL applicant is required to submit a comprehensive listing and evaluation of vital equipment, vital piping, vital power sources, vital water storage facilities, etc. The listing and evaluation must include all support functions vital to equipment operation (e.g., diesel generator (DG) cooling water; DG fuel; heating, ventilation, and air conditioning (HVAC) considerations for vital electrical equipment (DG); cooling water for component cooling). COL applicants must identify in their security plans equipment to be protected as vital. However, for a standard design, the staff expects that most of this list of vital equipment would not be site specific, and, therefore, should be included in the design certification application. The CESSAR-DC did not contain a complete, clearly identifiable listing of vital equipment. Requiring that a System 80+ standard set of vital systems be included in CESSAR-DC's vital equipment and vital area listing was designated as DSER Open Item 13.6-2. The staff's position and resolution is discussed in Section 13.6.2 of this chapter.

In CESSAR-DC Section 13.6.1(C), ABB-CE requires an evaluation to ensure that security system access restrictions do not interfere with safety, considering all modes of plant operation. The requirement did not explicitly require consideration in this evaluation of an access control system failure, such as a security power failure or a local card reader failure. Requiring consideration of impact of access control system failures in COL applicants' security system evaluations was designated as DSER Open Item 13.6-3.

In CESSAR-DC Amendment Q, ABB-CE added information requiring the COL applicant's design of the security system to include an evaluation of access control system failures. The staff considers this appropriate and DSER Open Item 13.6-3 is resolved.

In RAI Q500.17, the staff expressed its concern that the CESSAR-DC requires significant component compartmentalization for control of access to safety equipment, including single controlled access points to each channel equipment room. By letter dated December 17, 1991, ABB-CE responded:

The intention is to allow the Owner-Operator to incorporate subcompartmentalization techniques to protect vital equipment (using barriers provided for other purposes such as fire and flood protection) or to protect larger areas or groups of components, as may be necessary to meet the site-specific security program in a balance with maintenance functions. Several options would be available for emergency over-ride of failed security components.

Although ABB-CE's position, as stated in its reply to the RAI, is consistent with existing regulations, the staff concluded that ABB-CE should amend CESSAR-DC to delete possible contradictions to that position and to provide clarification on the proper balance that ensures that safety considerations are given priority. For example, the statements in Section 2.2 of CESSAR-DC Appendix 13A, that "Type I areas shall be minimized," and in Section 6.0 (I) of Appendix 13A, that physical access control separates each channel of the safety systems, may be too limiting in allowing COL applicants to address potential concerns regarding security system interference with emergency procedures. This was designated DSER Open Item 13.6-4. In CESSAR-DC Amendment Q, ABB-CE amended the CESSAR-DC guidelines for component compartmentalization for control of access to safety equipment. This revision permits the COL applicant to incorporate subcompartmentalization techniques to protect vital equipment or protect larger areas, as may be necessary to meet the site-specific security program. The revised sections eliminated the noted possible contradictions to this position and did provide clarification that safety considerations are given priority with respect to security considerations. The staff considers this appropriate and DSER Open Item 13.6-4 is resolved.

In CESSAR-DC Section 13.6.1(D), ABB-CE requires that security system facilities and equipment be located within the protected area.

In CESSAR-DC Section 13.6.1(E), ABB-CE requires security officers to have equipment for continuous commu-

nication with the alarm stations and the plant control room to have dedicated communications to the central alarm system (CAS) and the secondary alarm system (SAS). The staff was not satisfied that ABB-CE had adequately addressed ensuring the transmissibility of radio signals as well as equipment shielding to prevent radios from interfering with plant equipment. This issue was designated as part of DSER Open Item 9.5.2-3. The staff's resolution of this issue is discussed in Section 9.5.2 of this report.

The staff intends to review satisfactory compliance with interface requirements 13.6.1(A) through (E), as resolved in the final safety analysis report, during its review of COL applications referencing the System 80+ standard design. These reviews will be part of the site-specific review of security, contingency, and guard training plans required by 10 CFR 50.34 and 10 CFR Part 73.

13.6.1 Description of Sabotage Protection

In CESSAR-DC Appendix 13A Section 1.0, "Introduction," ABB-CE states that specific design criteria were developed to be consistent with the sabotage protection requirements of 10 CFR Part 73, as well as to give additional protection against a knowledgeable insider. These criteria were then implemented in the design as it proceeded. Sabotage resistance is described as comprising prevention, which is provided by control of access to vital equipment, and mitigation, which is provided by features to minimize the consequences of an event, should prevention fail. ABB-CE reviewed sabotage by considering the extent of diversity and defense in depth available for "certain key sabotage scenarios."

In CESSAR-DC Appendix 13A Section 2.1, "General Sabotage Design Assumptions and Criteria," ABB-CE describes the following as "general design criteria:"

- A. The knowledgeable insider has no weapons or explosives.
- B. The intrusion detection system cannot be covertly disabled.
- C. Prevention and mitigation systems are designed assuming an insider can initiate an event and disable multiple safety and non-safety systems, unless precluded by design features or prevented by plant security.
- D. Security resources are effective in preventing outsiders from assisting in insider sabotage.
- E. Equipment inside the containment is inaccessible to a saboteur during power operations.

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- F. Although the control room will be protected as a vital area in accordance with 10 CFR Part 73, the continued presence of multiple employees precludes an act of sabotage in the control room.
- G. During unit shutdown, a security area can be devitalized if justified in the physical security plan.
- H. Other single failures or initiating events do not have to be considered to occur coincidentally with sabotage events.
- I. Security system design generally needs to meet 10 CFR Part 50 (Appendix B) quality assurance standards.
- J. Security access controls must be compatible with access needed for fire protection, health physics, and actions operators take to mitigate events, considering loss of site power. Operator functions during normal operating modes should not be excessively impeded.
- K. The site security plan developed by the COL applicant will list all vital areas and equipment and identify the security systems protecting these areas and equipment.
- L. The site security plan developed by the COL applicant will include an evaluation of the security systems impact on plant operations, testing and maintenance during all modes of operations.

In CESSAR-DC Appendix 13A Section 2.2, "Access Control Design Criteria," ABB-CE describes the following as "plant layout and access control criteria:"

- A. Plant layout should allow access to all vital areas without requiring passage through a different vital or non-vital area.
- B. Protection of vital areas should be prioritized according to rankings of systems and components.
- C. The plant layout should include space for security equipment.
- D. Outdoor locations within the protected area to be protected against sabotage include: HVAC outside air suction for such inhabited spaces as the control room, CAS and SAS; diesel fueling ports and vents; diesel fuel storage and delivery systems; diesel cooling water systems; intake cooling water structures and systems; ultimate heat sink systems and structures; atmospheric steam dump valves and steam condenser bypass valves and controls; steam generator safety valves; auxiliary or startup transformers or both.

- E. Support systems for vital systems should be protected to the same degree as the served system.
- F. Penetrations through vital area boundaries should have appropriate intrusion barriers.
- G. Layout guidelines for separating and compartmentalizing equipment do not apply inside the containment.
- H. Control room layout should keep routine administrative activities not requiring contact with control room operators outside the control complex security boundary.
- I. Control room design will include bullet resistance requirement of 10 CFR 73.55.

In CESSAR-DC Appendix 13A Section 3.0, "Approaches to Access Control," ABB-CE's initial submittal described the options for access control assuming that positive identification is required to obtain access, so that tailgating is not possible. Access control options are described as: team zoning (e.g., two-man rule); area zoning, in which vital systems are segregated in separate vital areas and "access to opposing redundant system or component trains is controlled on a team basis, where one team has access to one train"; operational zoning (access to a component is restricted until its redundant component is demonstrated operable); time zoning (access is restricted to certain time periods when sabotage is less likely or more easily detected); and function zoning (activities personnel can perform are restricted). In Amendment Q, ABB-CE amended CESSAR-DC to remove the reference to various zoning access control options. The revised sections state the design was developed in a manner which permits and supports the deployment of several access control approaches. The design allows the COL applicant to determine the appropriate access control for the specific site requirements. In Section 3.0 above, ABB-CE stated that the flexibility of the design and layout can support various access control approaches.

In CESSAR-DC Appendix 13A initial submittal, Section 4.0, "System 80+ Standard Design Sabotage Protection Strategies and System Rankings," ABB-CE described revising the Sandia National Laboratory Report SAND82-7053, "Ranking of Light Water Reactor Systems for Sabotage Protection," to apply to the System 80+ design. Four other protection strategies were described, each of which is intended as a strategy for assuring the availability of a minimal but sufficient set of equipment to mitigate insider-induced transients and to either prevent or mitigate insider-induced loss-of-coolant accidents. Importance of the various plant systems was ranked according to how fast they would be required to function following a sabotage

attempt; that is, whether automatic system actuation was necessary or whether there was time for manual actions, such as manually starting a pump or realigning a backup system. ABB-CE deleted this section when the sabotage vulnerability analysis contained in its revised Section 7 entitled, "System 80+ Standard Design Sabotage Protection Strategies" was updated by Amendment T. The CESSAR-DC instead uses the probabilistic risk assessment (PRA) insights for determining the susceptibility to sabotage and minimal set of equipment for each protection strategy. Section 7 describes the protection strategies for the System 80+ design.

In CESSAR-DC Appendix 13A initial submittal, Section 5.0, "Evaluation of Damage Control Measures," ABB-CE evaluated 27 potential design modifications, referred to as damage control measures (DCMs), against 8 specific scenarios involving sabotage of certain components. This information was placed in Appendix 13A Section 8 and Table 13A-1. In Section 8.3, ABB-CE states that evaluating the DCMs against these scenarios revealed that the System 80+ design already incorporates the necessary DCMs to mitigate the scenarios.

In CESSAR-DC Appendix 13A initial submittal, Section 6.0 (revised Section 4.0), "Design Features for Sabotage Resistance," ABB-CE described System 80+ features represented as improving inherent resistance of the design to sabotage over traditional designs. These features include redesign of the emergency feedwater system (EFWS), with two emergency feedwater storage tanks located inside the auxiliary building. The refueling water storage tank is located inside the containment. The safety injection system (SIS) can function at higher pressures. The Nuplex 80+ instrumentation and controls design includes on-line monitoring and testing features making detection of sabotage attempts more likely.

In CESSAR-DC Appendix 13A revised Section 5.0, "Plant Layout for Sabotage Resistance," ABB-CE states that the plant layout follows the access control criteria of Appendix 13A Section 2.2. ABB-CE states that there is "significant component compartmentalization which can provide additional access control options, thereby permitting the deployment of a variety of access control strategies."

13.6.2 Evaluation

10 CFR 52.47 requires an application for design certification to contain the "technical information which is required of applicants for...operating licenses by...[10 CFR Part] 73..., and which is technically relevant to the design and not site-specific." In addition, in its "Policy Statement on Severe Reactor Accidents Regarding Future Designs and

Existing Plants," August 8, 1985, the Commission stated: "The Commission also recognizes the importance of such potential contributors to severe accident risk as human performance and sabotage. The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants." The discussion in CESSAR-DC Appendix 13A initially was deficient for the following three reasons:

(1) Vital Areas

Applicants for operating licenses must identify equipment to be protected as vital in their security plans. For a standard design, the staff expects that most of this list of vital equipment would not be site specific, and, therefore, should be included in the design certification application. The CESSAR-DC did not contain a complete, clearly identifiable listing of vital equipment. This was designated DSER Open Item 13.6-2.

In accordance with SRP Section 13.6 and Review Guideline 17 of the NRC Security Plan Evaluation Report Workbook, the staff considers that all seismic Category I systems should be designated as vital equipment in the sense of 10 CFR Part 73. By letter dated September 28, 1989, responding to RAI Q500.7, ABB-CE committed to revise CESSAR-DC Section 7.1.2.16.A to reference Appendix 13A and to update original Section 7 which has been incorporated into Section 5 of Appendix 13A to add:

"The following separate vital plant areas are defined for the System 80+ standard design:

- MCR
- remote shutdown control room
- Channel A equipment room
- Channel B equipment room
- Channel C equipment room
- Channel D equipment room"

"In addition, the following separate plant areas are also provided and contain restricted plant entry as well as measures to impede forced entry:

- computer room
- Channel X equipment room
- Channel Y equipment room"

"The Motor Control Centers (MCCs) are similarly located in physically separate rooms according to control channel assignment (A, B, C, D). Each room has a single controlled access point to restrict entry."

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In RAI Q500.21, the staff then requested the following:

"Considering the guidance of NRC Review Guideline 17 and Utility Requirements Document (URD) Chapter 9 Section 5.2.3, include in CESSAR-DC Section 13.6 a standard list of vital systems and vital areas for System 80+ to be incorporated into site-specific security plans in place of the list provided in Response 500.7. Provide a sound technical basis for any seismic Category I systems that would not be contained within vital areas."

By letter dated December 17, 1991, ABB-CE replied: "Section 12.3.1.3 of CESSAR-DC lists the control room, the TSC and the DG rooms as vital areas." This was not a complete list. The list of vital areas is addressed as part of DSER Open Item 13.6-2.

The staff notes that ABB-CE addresses guidance dealing with postaccident radiation levels in pathways to areas vital to postaccident operator activities (CESSAR-DC Section 12.3.1.3). These are not necessarily the same areas as those that would be designated as vital for the purpose of meeting the requirements of 10 CFR 73.55 for physical protection against radiological sabotage.

In addition to the vital areas listed above, ABB-CE's letter of February 28, 1992, included a proposed Section 1.2.1.4.1.5 (G), which specified that the combustion turbine generator (CTG) building and its related auxiliary structures will be designated vital areas, because the CTG provides backup security system power.

ABB-CE did not clearly identify a standard list of equipment that would be included in all COL applicant lists of vital equipment. In addition, the equipment that ABB-CE indicated would be identified in the CESSAR-DC as vital was incomplete; it did not include all seismic Category I equipment. The staff considers that the systems required to place the reactor in safe shutdown are vital systems; therefore, the seismic Category I SIS, reactor protection system, emergency DG and its fuel storage and transfer system, emergency power storage system, emergency onsite power distribution system, EFWS, atmospheric steam dump system, safety depressurization system, station service water system, and component cooling water system should be included in the System 80+ standard list of vital equipment. In addition, the staff considers that components necessary for protecting the integrity of the reactor coolant system (RCS) boundary should be considered vital equipment, including piping, valves, motor-control centers, and power distribution components necessary for isolating the RCS. Control stations for vital equipment, including the MCR and the remote shutdown panel (RSP), should also be in one or separate vital areas. In addition, the

spent fuel pool should be considered vital, as well as any other equipment that conforms to the definition of vital equipment in 10 CFR 73.2 and NRC Review Guideline 17. ABB-CE needed to submit an acceptable standard list of vital equipment, and develop a sound technical basis for any seismic Category I systems that would not be contained within a vital area. This was designated DSER Open Item 13.6-2. By Amendments Q and U, ABB-CE added CESSAR-DC Appendix 13B Tables 13B-1 through 13B-29 that listed vital equipment by major plant systems within the standard design scope. These tables containing seismic Category I equipment defined as vital equipment are consistent with NRC Review Guideline 17. On this basis, DSER Open Item 13.6-2 is resolved.

(2) Vulnerability Analyses

In its "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," August 8, 1985, the Commission stated: "The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants." In Volume II, Chapter 9, Section 5.2.2.1 (Rev. 1) of the URD, Electric Power Research Institute (EPRI) states: "To optimize system design and compatibility of plant arrangement and system design with sabotage protection, a sabotage vulnerability analysis shall be conducted prior to finalizing design." In its summary and recommendations on design and equipment, the staff noted in NUREG-1267, "Technical Resolution of Generic Safety Issue A-29, 'Nuclear Power Plant Design for Reduction of Vulnerability to Industrial Sabotage,'" (September 1987) the potential for insider access to and sabotage of redundant safety systems, particularly of "systems in standby mode that lack a status indicator in the control room," and recommended a systematic evaluation to identify these types of vulnerabilities and possible corrective actions.

In CESSAR-DC Appendix 13A Section 6.0 (revised Section 4.0), Item H, ABB-CE notes that the Nuplex 80+ instrumentation and controls design incorporates "on-line monitoring of fluid and electrical systems making detection of sabotage attempts more likely." In RAI Q500.25(b), the staff asked if this system would enable operators to detect mispositioning of manual isolation valves in these systems, and, if not, ABB-CE was asked to discuss the timeliness of discovery should a locked valve outside the containment be either mistakenly or deliberately mispositioned. ABB-CE's February 18, 1992, response to this RAI failed to address timeliness of discovery. Timely detection of mispositioned manual valves was designated DSER Open Item 13.6.2-1. In Amendment T, ABB-CE updated the list of valves in CESSAR-DC Appen-

dix 13A Section 6 with position indications in the control room. The NRC staff's review of the list noted that it contained the minimum number of valves with indication in the control room for vital systems. The additional valve position indication has enhanced the plant design to detect mispositioning of manual isolation valves in a timely manner. On this basis, DSER Open Item 13.6.2-1 is resolved.

In the discussion of sabotage protection in Appendix 13A revised Section 8.0, ABB-CE analyzes potential design alternatives but failed to examine the proposed design for potential sabotage vulnerabilities that perhaps could be eliminated by practical design changes. Neither the sabotage ranking analysis nor the DCM analysis in Appendix 13A considered:

- vital components and addressed whether or not they would be susceptible to tampering by an insider with authorized access
- whether or not timely means would be available to discover and compensate for tampering
- whether there were any practical design or procedure changes that could be made to decrease or eliminate opportunities for tampering or to ensure the ability to compensate for tampering

Performance of an insider vulnerability analysis was designated as DSER Open Item 13.6.2-2.

In Amendment T, ABB-CE revised Appendix 13A to incorporate the review for vulnerabilities to sabotage. The review used the System 80+ PRA as the basis for determining the vulnerabilities to sabotage. Each dominant accident sequence was reviewed to identify which sequences could be influenced by an act of sabotage.

For each accident sequence that could be affected, a strategy was developed to limit the effect. The strategy included a review of the design features with respect to limiting the effects of acts of sabotage by considering design enhancements or additional detection of sabotage. Section 7.0 of Appendix 13A describes the results of this review for each dominant accident sequence. From this review, ABB-CE incorporated additional valves that will be alarmed in the control room and would enhance the ability to detect valve mispositions in critical systems. These valves are listed in Section 6.0 of Appendix 13A. The final design, however, will still have to rely on site-specific installed security systems. As noted in Section 13.6.1 F of CESSAR-DC, a site-specific vulnerability analysis is required by the COL applicant to ensure that significant changes do not occur in the sabotage effects on

each dominant accident sequence and, where practical, attempt to optimize the specific site system arrangements and design with respect to sabotage protection before finalizing the design. The staff intends to review the site-specific vulnerability analysis and final design during its review of the COL site-specific security, contingency, and guard training plans required by 10 CFR 50.34 and 10 CFR Part 73.

In its February 18, 1992, response to RAI Q500.32, ABB-CE described alarms that would indicate to operators in the MCR when the RSP room is accessed. ABB-CE stated that the four transfer switch rooms previously shown in CESSAR-DC Figure 9.5.1-3 to be in close proximity to the RSP have been eliminated. In Chapter 10, Section 4.9.3.8 of the URD, EPRI stated:

Any action taken to enable a remote shutdown station or transfer control to it shall be annunciated in the MCR. The man-machine interface system designer shall make an assessment, using analysis and active simulation (including appropriate assumptions regarding support from security personnel), that the annunciation in the MCR provides adequate time for the operators and plant security to take action to prevent a serious accident in the case of unauthorized use of a remote shutdown station.

With revision to the locations of the transfer switches, ABB-CE considered that unauthorized transfer of all control to the RSP is not practical. An adversary outside the MCR cannot prevent reactor trip by operators because the manual reactor trip pushbuttons in the MCR are never disabled. Controls that can be transferred from outside the MCR require access to five channelized instrumentation and control (I&C) equipment rooms and the RSP in separate locations. Alarms are generated in the MCR when a door to the RSP or to any of the I&C equipment rooms are opened. In addition, alarms are generated in the MCR if any of the channel transfers is actuated. The staff agrees that unauthorized transfer of all control to the RSP is not practical and finds ABB-CE's response acceptable.

(3) Safety/Safeguards

ABB-CE paid insufficient attention to possible security system interference with safe operation of the plant. The access control design criterion initially described in Appendix 13A Section 2.2, stated that Type I areas should be minimized, and the approaches to access control presented in Section 3.0, may result in overcompartmentalization. The staff was concerned that overcompartmentalization could hinder rapid operator access to safety equipment.

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In its response to RAI Q500.17, ABB-CE proposed several modifications to Appendix 13A, including adding the following to Section 2.1:

- L. The site-specific security plan developed by the site operator shall include an evaluation of the security systems impact on plant operation, testing, and maintenance, considering all operational modes and/or emergency conditions.

However, in its response to RAI Q500.7, ABB-CE proposed amendments that would provide each I&C equipment room and each MCC with "a single controlled access point to restrict entry and incorporate measures to impede forced entry." Measures to impede forced entry implies a locked door at each point of access to equipment. In Chapter 9, Section 5.2.4.1 (Rev. 3) of the URD, EPRI specifies that security doors will not be required for controlling personnel access between the divisions of vital components.

ABB-CE amended CESSAR-DC to clarify that the System 80+ plant should be protected against sabotage without relying on locked security doors between adjacent vital equipment. This was designated as part of DSER Open Item 13.6-4. In CESSAR-DC Amendments Q and T, ABB-CE revised Sections 1 and 3 to enable the COL applicant to incorporate subcompartmentalization techniques as necessary to meet the site-specific security program in a balance with normal and emergency plant operational needs.

Protection for the seismic Category I SIS as described in CESSAR-DC Sections 6.3.1.2.3 and 6.3.2.6, did not address protection against sabotage. The staff considers that the SIS is a vital system; therefore, as required by 10 CFR 73.55 (c), access to all SIS components, including MCCs and electrical buses, should require passage through two barriers. (Locked security doors controlling access between two adjacent vital areas are not desirable, if access to each vital area is otherwise controlled.) Vital designation of the SIS system was included in DSER Open Item 13.6-2. By Amendment U, ABB-CE designated the SIS as a vital system in CESSAR-DC Section 6.3.1.2.3. The inclusion of vital equipment list as described in CESSAR-DC Appendix 13B and classification of SIS as a vital system resolves this item.

14 VERIFICATION PROGRAMS

14.1 Preliminary Safety Analysis Report Information

Section 14.1 of a standard safety analysis report is specified in Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports," and the standard review plan (SRP). However, the preliminary safety analysis report information specified by RG 1.70 is not applicable to either a final design approval (FDA) or a design certification under 10 CFR Part 52, and therefore, no information is contained in this section of the CESSAR-DC.

14.2 Initial Test Programs

The staff reviewed CESSAR-DC Section 14.2, "Specific Information in Final Safety Analysis Reports," and the applicant's response of February 18, 1992, to the NRC's request for additional information (RAI), in accordance with Section 14.2 of the Standard Review Plan, and RG 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants." SRP Section 14.2 applies to operating license applications. Therefore, some areas of review, such as an administrative manual that a combined license (COL) applicant would use to delineate plant operational conditions at which tests are conducted have not been evaluated. However, the staff identified these issues as COL action items that have been incorporated into the CESSAR-DC. These will be addressed by the COL applicant.

In CESSAR DC Section 14.2, ABB-CE describes the initial test program (ITP) for the System 80+ plant. The ITP consists of a series of tests categorized as prerequisite tests, preoperational tests, fuel loading and post-core hot functional tests, initial criticality and low-power physics tests, and power ascension tests. The prerequisite tests determine proper installation and functional operability of equipment. Preoperational tests are those tests normally conducted before fuel loading to demonstrate that plant systems operate in accordance with the design. Fuel loading and post-core hot functional tests offer a systematic approach for safely accomplishing and verifying the initial fuel loading and offer additional assurances that plant systems function as expected. Initial criticality and low-power physics tests ensure that criticality is achieved in a safe and controlled manner and verify selected core design parameters. Power ascension tests demonstrate that the integrated plant operates in accordance with the design. This section documents the staff's evaluation of the System 80+ ITP, including the ABB-CE responses to the staff's RAI.

The ITP described in CESSAR-DC Section 14.2 offers sufficient detail with respect to all test descriptions to

enable the staff to determine the adequacy of all the test abstracts. The staff concludes that the described ITP is generally comprehensive, answers all the RAI questions and all the open and confirmatory items from the draft safety evaluation report (DSER), and covers all areas of staff concern. The evaluation includes the resolution of the CESSAR-DC sections that the staff initially found needed modifications and additional information.

The section numbers in this Chapter correspond to those in the CESSAR-DC, as closely as possible.

14.2.1 Test Program and Objectives

The ITP includes testing activities commencing with the completion of construction and installation, and ending with the completion of the power ascension testing. This test program demonstrates that components and systems operate in accordance with design requirements and meet the requirements of SRP Section 14.2. It also aids in the establishment of baseline performance data and serves to verify that normal operating and emergency procedures accomplish their intended purposes. The ITP consists of prerequisite testing plus the following four phases:

- Phase I: Preoperational Testing
- Phase II: Fuel Loading and Post-Core Hot Functional Testing
- Phase III: Initial Criticality and Low-Power Physics Testing
- Phase IV: Power Ascension Testing

Prerequisite testing consists of tests and inspections required to ensure construction is complete and that systems are ready for Phase I testing.

Phase I demonstrates that structures, systems, and components (SSCs) operate in accordance with design operating modes throughout the full design operating range. Where required, simulated signals or inputs are used to demonstrate the full range of the systems that are used during normal operation. Systems that are not used during normal plant operation, but that must be in a state of readiness to perform safety functions, are checked under various modes and test conditions before fuel is loaded. Preoperational testing ensures that systems and equipment perform in accordance with the safety analysis report. Test results are analyzed to verify that systems and components are performing satisfactorily, and if they are not, to provide a basis for recommended corrective actions. Upon completion of the specific preoperational testing, a series of integrated system tests, typically termed "pre-core hot functional testing," is performed to verify proper systems operation before fuel is loaded.

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Phase II is a systematic process for safety accomplishing and verifying the initial fuel loading. The post-core hot functional tests are performed between initial fuel loading operations and initial criticality to verify that plant systems necessary for normal plant operation function as expected and to obtain performance data on core-related systems and components.

Phase III of the startup test program assures that criticality is achieved in a safe and controlled manner. After initial criticality is achieved, a series of low-power physics tests are performed to verify selected core design parameters. These tests substantiate the safety analysis and technical specifications. They also demonstrate that core characteristics are within expected limits and produce data for benchmarking the design methodology used for predicting core characteristics later in life.

Phase IV brings the reactor to full power. Testing is performed at plateaus of approximately 20-, 50-, 80-, and 100-percent power to demonstrate correct performance of the facility during steady-state conditions and such anticipated transients as reactor trips (scrams) from full power and loss of offsite power.

The objectives for the ITP stated in CESSAR-DC Section 14.2.1 conform to the applicable parts of SRP Section 14.2.

14.2.2 Organization and Staffing

In CESSAR-LC, ABB-CE adequately describes the typical licensee's organization and staffing for a System 80+ plant. The management organization designates a senior level manager who has overall responsibility for defining the responsibilities, requirements, and working relationships necessary to safely and efficiently design, construct, start up, operate, maintain, and modify the nuclear power plant. This manager is assisted by other senior level managers responsible for developing test procedures and other related documentation for the ITP. ITP procedures are developed by the Operations Department with assistance from the nuclear steam supply system vendor, the architect-engineer, and other vendors as required. The startup organization comprises system engineers who have assigned responsibility for specific systems and startup engineers who have responsibility for testing activities and specific tests. Other groups assisting in the development of the ITP include the Test Working Group which gives advice on the technical adequacy of the ITP and the Plant Review Board which provides high-level review and approval of the ITP. In accordance with the review guidance provided in SRP Section 14.2 and RG 1.68, Revision 2, the staff finds that the description of the site organization and staffing is acceptable. The COL applicant

is responsible for developing the specific organization and staffing levels appropriate for its facility. The plant organizational staff will be utilized to the extent practicable during the conduct of the ITP.

14.2.3 Test Procedures

In CESSAR-DC, ABB-CE describes the methods the COL applicant can use for preparation and organizational approval of prerequisite test procedures, Phases I through IV test procedures, and special test procedures. The staff concludes that the methods described for preparing and approving test procedures are acceptable.

The COL applicant is responsible for assuring the preparation and designation of the approval process for prerequisite and Phases I and IV test procedures. The COL applicant will also submit preoperational and startup test summaries that contain testing objectives and acceptance criteria applicable for its scope of the plant design. The COL applicant will need to submit the following for NRC staff review:

- (1) Phase I preoperational test procedures approximately 60 days before their intended use and Phases II through IV startup test procedures approximately 60 days before fuel loading. This is COL Action Item 14.2.3-1.
- (2) the scoping documents (i.e., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to its scope of design responsibility. This is COL Action Item 14.2.3-2.
- (3) the startup administrative manual (procedures) and any other documents that delineate plant operational conditions where tests are to be conducted, testing methodologies to be utilized, specific data to be collected, and acceptable data-reduction techniques to be reviewed by the staff at the time of the combined license. This is COL Action Item 14.2.3-3.
- (4) any reconciliation methods needed to account for test conditions, methods, or results if testing is performed at condition other than representative design operating conditions. This is COL Action Item 14.2.3-4.

14.2.4 Conduct of Test Program

In CESSAR-DC, ABB-CE describes a general methodology for carrying out test procedures once they have been cleared for performance. This includes descriptions of

such plant personnel as the test director, the startup engineer, the operating shift supervisor, and other plant testing staff responsible for conducting the test program in a safe manner. ABB-CE also discusses methods to control signoff provisions, maintenance and modification procedures, and test performance.

In accordance with the review guidance provided in SRP Section 14.2 and RG 1.68, Revision 2, the staff concludes that the provisions described by ABB-CE for conducting the test program are acceptable. The COL applicant is responsible for planning the plant startup program appropriate for its facility. This is COL Action Item 14.2.4-1.

14.2.5 Review, Evaluation, and Approval of Phases I Through IV Test Results

In CESSAR-DC, ABB-CE states that test results for each phase of the test program should be reviewed and verified to be complete and satisfactory before the next phase is started. This is the responsibility of the COL applicant. Preoperational testing on a system will not normally begin until all applicable prerequisite tests have been completed, reviewed, and approved. Before initial fuel loading and the commencement of initial criticality, the Test Working Group will conduct a comprehensive review of required completed preoperational procedures. This review will ensure that required plant systems and structures will be capable of supporting the initial fuel loading and subsequent startup testing.

It is intended that Phase I testing be completed before commencing initial fuel loading. If prerequisite and Phase I testing are incomplete at initial fuel loading, provisions for carrying over testing will be planned and approved in accordance with the site-specific administrative procedures.

The staff asked ABB-CE to modify CESSAR-DC Section 14.2.5 to state that Phase I testing (including the review and approval of the test results) must be completed before fuel loading and, if portions of any preoperational tests are intended to be conducted, or their results approved, after fuel loading, to require the COL applicant referencing the System 80+ design certification to (1) list each test, (2) state which portions of each test will be delayed until after fuel loading, (3) give technical justification for delaying these portions, and (4) state the power level at which each test will be completed.

ABB-CE revised CESSAR-DC Section 14.2.5, "Review, Evaluation, and Approval of Phases I Through IV Test Results," by adding the following paragraph:

A listing will be provided in the Phase I Test Results Report stating which tests or portions of a

test will be delayed until after fuel loading. A technical justification will be provided along with the anticipated power level/mode [at which] the test will be performed. This will be approved by the plant review board as stated previously in Section 14.2.2.8.

The staff found this change unacceptable and asked ABB-CE to clearly acknowledge in CESSAR-DC Section 14.2.5 or CESSAR-DC Section 14.2.11 that Phase I testing (including the review and approval of the test results) is required before fuel loading. The staff concluded that any preoperational tests or portions of these tests to be carried over after fuel load must be identified, and appropriate justification must be given during the application process. It is not appropriate to make such changes to the ITP during the conduct of preoperational testing. This was designated as DSER Open Item 14.2.5-1.

ABB-CE revised CESSAR-DC Section 14.2.5 in Amendment L. The staff reviewed the revision and noted that all Phase I preoperational and prerequisite testing would be completed before fuel would be loaded and that any required carryover testing will be identified, justified, approved, and made available for NRC review before commencement of fuel loading. In accordance with the review guidance provided in SRP Section 14.2 and RG 1.68, Revision 2, the staff finds this change acceptable. On this basis, DSER Open Item 14.2.5-1 is resolved.

14.2.6 Test Records

The operator of the plant is responsible for retaining the test records for the life of the plant. This is COL Action Item 14.2.6.1.

14.2.7 Conformance of Test Program With Regulatory Guides

The ITP is consistent with the recommendations of the following regulatory guides associated with startup (with exceptions as noted and revisions as specified in CESSAR-DC Table 1.8-1): 1.9, 1.18, 1.20, 1.30, 1.41, 1.52, 1.68, 1.68.2, 1.68.3, 1.79, 1.108, 1.116, 1.118, and 1.140.

The staff had determined, however, that the applicant needed to address the following specific issues:

- (1) The exception to RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 2, Appendix A, Section 5.a, should have been revised to include testing at 20-percent and 80-percent power for the first-of-a-kind plant. For follow-on plants, reduced testing (50-percent, 100-

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percent) is appropriate only if more-stringent requirements are met for these conditions.

ABB-CE stated that the System 80+ plant is considered a "follow-on" plant for which testing at 50-percent and 100-percent power, to determine that power reactivity coefficients are in accordance with the design safety analysis, is acceptable only if more-stringent requirements are met for these conditions.

The staff determined that the applicant should have described these more-stringent requirements in the appropriate test abstract(s). This was designated as DSER Open Item 14.2.7-1.

By reviewing the applicant's revision to CESSAR-DC Section 14.2.7.1.4, "Reference Appendix A, Section 5.a" (Amendment L), the staff verified that reduced power reactivity coefficient testing for follow-on plants can be performed at 50-percent and 100-percent power levels if measurements of temperature reactivity coefficients at essentially zero reactor power are within the acceptance criteria established for follow-on plants. Reactivity coefficients are given in CESSAR-DC Table 14.2-6, "Physics (Steady State) Test Acceptance Criteria Tolerances." The staff concludes that the power and temperature reactivity coefficient acceptance criteria tolerances in CESSAR-DC Table 14.2-6 are acceptable. In accordance with the review guidance provided in SRP Section 14.2 and RG 1.68, Revision 2, the staff concludes that the changes to CESSAR-DC Section 14.2.7.1.4 (including the reactivity coefficients and tolerances) are acceptable. On this basis, DSER Open Item 14.2.7-1 is resolved.

- (2) RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," should have been listed among the applicable guides in accordance with SRP Section 14.2.

ABB-CE stated their intent to revise CESSAR-DC Section 14.2.7 to address RG 1.95. The staff reviewed the proposed changes and determined they were acceptable subject to incorporation into a CESSAR-DC amendment. This was designated as DSER Confirmatory Item 14.2.7-1.

The staff reviewed CESSAR-DC Section 14.2.7, Amendment L, and verified that RG 1.95 has been added to the list of applicable regulatory guides. The staff finds this change acceptable. On this basis, DSER Confirmatory Item 14.2.7-1 is resolved.

- (3) RG 1.139, "Guidance for Residual Heat Removal," should have been listed among the applicable guides in accordance with SRP Section 14.2.

ABB-CE stated their plan to revise CESSAR-DC Section 14.2.7 to address RG 1.139.

The staff reviewed the proposed changes and determined they were acceptable subject to incorporation into a CESSAR-DC amendment. This was designated as DSER Confirmatory Item 14.2.7-2.

The staff reviewed CESSAR-DC Section 14.2.7, Amendment L, and verified that RG 1.139 has been included in the list of applicable regulatory guides. The staff finds this change acceptable. On this basis, DSER Confirmatory Item 14.2.7-2 is resolved.

14.2.8 Utilization of Reactor Operating and Testing Experience in the Development of Initial Test Program

ABB-CE will continually send feedback to its startup organization during the development and execution of the ITP. This information reflects both ABB-CE operating and test experience and industrywide experience concerning pressurized-water reactors.

This review is performed by circulating licensee event reports (LERs) or summaries of LERs and NRC bulletins, circulars, and information notices to appropriate personnel so that pertinent information can be utilized in the startup program.

The utilization of reactor operating and testing experience in the development of the test program complies with Section 14.2 of the SRP.

14.2.9 Trial Use of Plant Operating and Emergency Procedures

In the CESSAR-DC, ABB-CE states that the COL applicant will develop a schedule for the development of plant operating and emergency procedures. This will allow sufficient time for trial use of these procedures during the ITP. The objectives for using the operating and emergency procedures during the ITP are to familiarize the plant operating and technical staff with the operation of the facility and to verify by trial use, to the extent practical, that the plant operating and emergency procedures are adequate. The COL applicant will need to develop a method to accomplish these objectives. This is COL Action Item 14.2.9-1.

14.2.10 Initial Fuel Loading and Initial Criticality

ABB-CE described the methods the COL applicant will use to control initial fuel loading and initial criticality. ABB-CE also describes the safe loading criteria, fuel loading procedures, and the safe criticality criteria. In accordance with the review guidance provided in SRP Section 14.2 and RG 1.68, Revision 2, the staff concludes that the description of the methods that the COL applicant will use to control initial fuel loading and initial criticality is acceptable. Overall direction, coordination, and control of the initial fuel loading and initial criticality activities are the responsibility of the COL applicant. ABB-CE will provide any technical assistance needed during the initial fuel loading and initial criticality activities. The COL applicant will need to prepare a startup administrative manual (procedures) and any other documents that delineate the methods of controlling prefuel load checks, initial fuel loading, precritical testing, and initial criticality. The NRC staff will review this information before issuing a combined license. This is COL Action Item 14.2.10-1.

14.2.11 Test Program Schedule

The test program schedule for plant startup will be developed by the COL applicant to allow sufficient time to systematically perform the required testing for Phases I through IV testing. Phase I test procedures are scheduled to be approved and available for NRC review approximately 60 days before their intended use. Phases II through IV test procedures are scheduled to be approved and available for NRC review approximately 60 days before fuel loading. In accordance with the review guidance provided in SRP Section 14.2 and RG 1.68, Revision 2, the staff concludes that this is acceptable. The COL applicant will also need to prepare a startup administrative manual and other documents that delineate the test program schedule for staff review. This is COL Action Item 14.2.11-1.

14.2.12 Test Abstracts

14.2.12.1 Individual Test Descriptions

In CESSAR-DC Section 14.2.12 and in a letter dated February 18, 1992, responding to an NRC request for additional information, ABB-CE submitted an abstract of the individual preoperational tests, post-core hot functional tests, low-power physics tests, and power ascension tests.

The staff asked ABB-CE to modify CESSAR-DC Section 14.2.12 so that the test abstracts address the following concerns:

- (1) Preoperational test prerequisites include the requirement that support systems required for system

testing are complete and operational. This level of detail in the test abstracts is insufficient to determine conformance with RG 1.68, Position C.2. The staff asked the applicant to address generic support system requirements in CESSAR-DC Section 14.2.12.1, and advised that the individual test abstracts should address specific support system requirements.

In CESSAR-DC Section 14.2.12.1, ABB-CE states that the test descriptions (scenarios) are test guides or abstracts that identify the scope of testing. They are not intended to give detailed information on the conduct of each test. The test guidelines developed from these abstracts will produce the level of detail necessary to describe (identify) more "specific" support system requirements. People who have appropriate technical background and experience will develop detailed preoperational test procedures from these guidelines. Therefore, support system wording will remain as is. Those tests that do not address construction activities completely (to the degree that outstanding construction items could not be expected to affect the validity of test results) will be modified in the next revision of CESSAR-DC Chapter 14 in order to support RG 1.68, Position C.2.

In the DSER, the staff asked ABB-CE to identify support systems required to be available for designated test phases as prerequisites for that test phase, and additional support systems required to be available for individual tests as prerequisites for these tests. This was designated as DSER Open Item 14.2.12.1-1.

By reviewing ABB-CE's response dated January 26, 1993, the staff verified that the preoperational test abstracts identified general prerequisites. The staff also verified that the preoperational testing sequence ensures that required supporting systems are available and operable to the extent necessary to perform a specific test. Upon completion of preoperational testing, plant systems are essentially operational and, consequently, available to support the subsequent post-core hot functional, low-power physics and power ascension testing. ABB-CE's response also stated that detailed test procedures would summarize specific prerequisites based on the associated test methodology. ABB-CE also submitted CESSAR-DC Table 14.2.7, "Matrix of Support Systems Recommended for Preoperational Tests," to identify the major support systems (not all inclusive) required for each preoperational test.

The staff concludes that test prerequisites for support systems contained in CESSAR-DC Section 14.2.12.1, "Preoperational Tests," should also be incorporated into

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CESSAR-DC Table 14.2.7. The individual preoperational test abstracts should also be modified to include the applicable prerequisite information from Table 14.2.7. The applicant responded in a letter dated April 21, 1993, by appropriately clarifying Table 14.2.7, and the prerequisite information in Section 14.2.12.1. The revisions to Table 14.2.7 and Section 14.2.12.1 were subsequently incorporated into CESSAR-DC in Amendment O. The staff verified that the applicant included the appropriate test prerequisite requirement revisions in Section 14.2.12.1 and Table 14.2.7. Based on this verification, the staff concludes that the revisions are acceptable in accordance with the review guidance provided in SRP Section 14.2 and RG 1.68, Revision 2. On this basis, DSER Open Item 14.2.12.1-1 is resolved.

- (2) Several test abstracts contain imprecise acceptance criteria (e.g., "acceptable," "allowable," "anticipated," "design," "estimated," "expected," "manufacturer's instructions," "proper," "selected," "specified," "within assumed uncertainties," "within limits"). The staff asked the applicant to modify individual test abstracts to specify the bases for determining acceptable system and component performance. Acceptable criteria include specific references to regulatory guides, technical specifications, assumptions used in the safety analysis, other sections of the CESSAR-DC, and applicable codes and standards.

ABB-CE stated that the preoperational test acceptance criteria have been reviewed and the criteria identified in CESSAR-DC Sections 14.2.12.1, 14.2.12.2, 14.2.12.3, and 14.2.12.4 require revision. ABB-CE stated that CESSAR-DC Chapter 14 would be revised to reference test acceptance criteria compliance with regulatory guides, technical specifications, assumptions used in safety analysis, other CESSAR-DC sections, and applicable codes and standards. This was designated as DSER Open Item 14.2.12.1-2.

The staff reviewed the revised test abstracts submitted by ABB-CE and determined that several acceptable changes had been made, but that not all of the test abstracts were modified with the appropriate acceptance criteria information. The staff asked ABB-CE to modify the acceptance criteria information for the test abstracts listed below to specify the bases for determining acceptable system and component performance.

- Test 14.2.12.1.6, CVCS Purification Subsystem Test
- Test 14.2.12.1.9, Chemical Addition Subsystem Test
- Test 14.2.12.1.50, Remote Shutdown Panel
- Test 14.2.12.1.58, Pre-Core Reactor Coolant System Heat Loss

- Test 14.2.12.1.75, Feedwater Heater and Drains System Test
- Test 14.2.12.1.91, Heat Tracing Systems Test
- Test 14.2.12.1.137, Mid-Loop Operations Verification Test
- Test 14.2.12.3.1, Low-Power Biological Shield Survey Test
- Test 14.2.12.4.5, Turbine Trip Test
- Test 14.2.12.4.6, Unit Load Rejection Test

By letter dated April 21, 1993, ABB-CE submitted a draft that included the appropriate cross-reference to other sections of CESSAR-DC that describe acceptance criteria in Chapter 14 for eight of the tests listed above. Adequate cross-references were not submitted for CESSAR-DC Sections 14.2.12.4.5, "Turbine Trip Test," and 14.2.12.4.6, "Unit Load Rejection Test." By letter dated May 4, 1993, ABB-CE submitted cross-references to acceptance criteria for these two tests in CESSAR-DC Chapter 15. All of the cross-references were incorporated into CESSAR-DC in Amendment O. The staff also found additional CESSAR-DC cross-references for various testing methods and their associated acceptance criteria for 37 preoperational and startup tests that needed to be incorporated into CESSAR-DC Section 14.2. These additional cross-references were sent in a letter to the applicant dated September 9, 1993. In response, ABB-CE submitted additional cross-references in CESSAR-DC Amendments T and U. The changes to CESSAR-DC are acceptable. On this basis, DSER Open Item 14.2.12.1-2 is resolved.

- (3) By letter dated February 18, 1992, ABB-CE submitted test abstracts in a response to RAI Q640.6. The staff reviewed the responses to verify that the test abstracts provided more detailed test acceptance criteria and found them acceptable in accordance with the review guidance provided in SRP Section 14.2 and RG 1.68, Revision 2. However, the CESSAR-DC needed to be updated to reflect ABB-CE's response. This was designated as DSER Confirmatory Item 14.2.12.1-2.

ABB-CE incorporated the test abstracts into the CESSAR-DC (Amendment J), and the staff finds this acceptable. The status of the acceptability of each of these test abstracts is the subject of DSER Open Item 14.2.12.1-1 and 14.2.12.1-2 indicated above, and also the DSER open items described below under Section 14.2.12.2 of this report. On this basis, DSER Confirmatory Item 14.2.12.1-1 is resolved.

The staff asked ABB-CE to identify startup tests listed in CESSAR-DC Section 14.2.12.2 that are not essential to the demonstration of conformance with design requirements

for structures, systems, components, and features, but that meet any of the following criteria:

- startup tests that will be used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period
- startup tests that will be used for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions
- startup tests that will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications
- startup tests that are classified an engineered safety features or will be used to support or ensure the operations of engineered safety features within design limits
- startup tests that are assumed to function or for which credit is taken in the accident analysis for the facility, as described for the System 80+ standard design
- startup tests that will be used to process, store, control, or limit the release of radioactive materials

ABB-CE identifies Startup Test 14.2.12.2.1, "Post-core Hot Functional Test Controlling Document," as the only test that belongs in this category since it is an administrative procedure used to direct the operation of the plant systems during the post-core load testing period. This item is acceptable subject to its incorporation into the CESSAR-DC. This was designated as DSER Confirmatory Item 14.2.12.1-2.

The staff later determined that the information concerning tests not essential to the demonstration of conformance with design requirements for SSCs but that meet one of the preceding criteria, was not required to demonstrate compliance with RG 1.68 (Rev. 2). The information was not needed for the staff to make a safety determination. On this basis, DSER Confirmatory Item 14.2.12.1-2 is resolved.

The staff also determined that both the pre-core and post-core hot functional tests controlling documents are administrative test procedures used to demonstrate the integrated operation of plant primary, secondary, and auxiliary

systems preceding initial criticality. Therefore, these administrative procedures are essential to the demonstration of conformance with design requirements for SSCs and features important to safety. These procedures also address the guidance of RG 1.68, Position C.4, "Test Procedures." The applicant responded in Amendment T by adding Section 14.2.7.1.13, "Test Procedures," which stated "Test summaries 14.2.12.2.1, Post-Core Hot Functional Test Controlling Document and 14.2.12.1.48, Pre-Core Hot Functional Test Controlling Document are essential to the demonstration of conformance with requirements for SSCs and features important to safety." Further discussions on clarifications to RG 1.68 commitments with respect to the use of the controlling documents were also given in CESSAR-DC Section 14.2.7. The staff also determined that more details on the controlling documents would be included in the startup administrative manual which will be supplied by the COL applicant. Therefore, the clarifications in CESSAR-DC Section 14.2.7 are acceptable.

14.2.12.2 Conformance of the CESSAR-DC With RG 1.68, Revision 2

The staff reviewed the preoperational and startup test phase descriptions and found that the operability of several of the systems and components listed in RG 1.68 may not be adequately demonstrated by the tests described in the CESSAR-DC.

The staff asked ABB-CE to either expand the test descriptions to address the following items, insert cross-references in CESSAR-DC Section 14.2.12 if complete test descriptions for the following items are given elsewhere in the CESSAR-DC, or modify CESSAR-DC Section 14.2.7 and CESSAR-DC Table 1.8-1, as appropriate, to provide technical justification for any exception to RG 1.68, Revision 2. Therefore, the following items should be reflected in a revision to the CESSAR-DC. (Note: Each item is numbered in accordance with RG 1.68, Revision 2.)

1.h.(4) Hydrogen mitigation systems (Revision to Sections 14.2.12.1.112 and 14.2.12.1.113)

ABB-CE revised CESSAR-DC Sections 14.2.12.1.112 and 14.2.12.1.113 in Amendment J to address tests and acceptance criteria for the hydrogen mitigation system and the containment hydrogen recombiner system. In accordance with the review guidance provided in SRP Section 14.2 and RG 1.68, Revision 2, Appendix A, Regulatory Position 1(h).(4), the staff finds the proposed revisions to the test abstracts and their acceptance criteria are acceptable.

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1.h.(5) Cold water interlocks

ABB-CE stated that cold water interlocks are not applicable.

ABB-CE was asked to identify this exception to RG 1.68, Appendix A, in CESSAR-DC Section 14.2.7, "Conformance of Test Programs With Regulatory Guides."

By letter dated April 21, 1993, ABB-CE submitted a draft of Section 14.2.7.1.9, which states "Cold Water interlocks are not applicable to the System 80+ design. This testing will not be performed as it is not applicable." This is acceptable. This information was incorporated into CESSAR-DC in Amendment O. On this basis, DSER Confirmatory Item 14.2.12.2-1 is resolved.

1.h.(8) Holdup volume tank and cavity flooding system.

ABB-CE revised CESSAR-DC Section 14.2.12.1.42 in Amendment J to include tests for the holdup volume tank and the cavity flooding system. The staff reviewed the new material; it is acceptable.

1.i.(2) Containment isolation valve functional and closure timing.

ABB-CE added Section 14.2.12.1.140 in Amendment J of the CESSAR-DC to include valve functional and closure timing tests. The staff reviewed the new material; it is acceptable.

1.i.(8) Primary and secondary containment isolation actuation system (CIAS) logic

ABB-CE stated that containment isolation actuation testing is performed as part of the engineered safety features component control system (ELF-11CCS) test, CESSAR-DC Section 14.2.12.1.25.

The staff has determined that CESSAR-DC Section 14.2.12.1.25 does include testing of the trip logic and actuation for the CIAS. This is acceptable.

1.i.(21) Containment penetration cooling system

ABB-CE stated that a containment penetration cooling system is not applicable because System 80+ does not require such a system.

The staff asked the applicant to identify this exception to RG 1.68, Appendix A, in CESSAR-DC Section 14.2.7, "Conformance of Test Programs With Regulatory Guides." This was designated as DSER Confirmatory Item 14.2.12.2-2.

By letter dated April 21, 1993, ABB-CE submitted a draft of Section 14.2.7.1.10, which stated "A containment penetration cooling system is not a design requirement for the CE System 80+. This testing will not be performed as it is not applicable." This information was incorporated into CESSAR-DC in Amendment O. This is acceptable. On this basis, DSER Confirmatory Item 14.2.12.2-2 is resolved.

1.j.(7) Leak detection systems used to detect failures in emergency core cooling system (ECCS) and containment recirculating spray systems located outside containment.

In Amendment J, ABB-CE revised CESSAR-DC Section 14.2.12.1.84 to address sump level instrumentation. Alarms and interlocks are tested and verified operable per design for the equipment and floor drainage system. These changes are acceptable.

1.j.(12) Failed fuel detection system.

In Amendment J, ABB-CE revised CESSAR-DC Sections 14.2.12.1.19 and 14.2.12.1.118 to include test objectives, prerequisites, and acceptance criteria for the letdown process radiation monitor system and acceptance criteria for the airborne and area radiation monitors. These changes are acceptable.

1.j.(20) Instrumentation used to detect external and internal flooding conditions

In Amendment J, ABB-CE revised CESSAR-DC Section 14.2.12.1.84 to address this issue. Objectives and test methods were revised to demonstrate that system segregation is maintained in the equipment and floor drainage system. These changes are acceptable.

1.j.(22) Instrumentation that can be used to track the course of postulated accidents.

In Amendment J, ABB-CE added CESSAR-DC Section 14.2.12.1.141, "Post Accident Monitoring Instrumentation Test," to address this issue. This change is acceptable.

1.k.(2) Personnel monitors and radiation survey instruments

ABB-CE stated that personnel monitors and radiation survey instruments are site-specific items to be addressed by the COL applicant.

The staff asked ABB-CE to identify this item as a site-specific item for which the COL applicant referencing the

System 80+ design certification will submit the required test abstract.

ABB-CE sent a letter to the NRC dated January 18, 1992, to address this as a COL applicant issue. ABB-CE proposed to add the following statement in their draft revision, "Personnel monitors and radiation survey instruments are site-specific items to be addressed by the COL applicant. The COL applicant will define the appropriate testing to demonstrate proper operation of personnel monitors and radiation survey instruments." This response is acceptable. This DSER open item created COL Action Item 14.2.12.2-1. ABB-CE incorporated this COL applicant information into the CESSAR-DC in Amendment L. On this basis, DSER Open Item 14.2.12.2-1 is resolved.

1.m.(8) Fuel handling equipment operability at 100-percent load.

In Amendment J, ABB-CE revised CESSAR-DC Section 14.2.12.1.99, "Fuel Handling Cranes Test," to address operability of cranes at 100-percent load. Additional test methods include a 150-percent static load test, a 100-percent rated load test, and verification on operability of protective and safety devices. This is acceptable.

1.n.(8) Seal water systems

ABB-CE stated that the reactor coolant pump (RCP) seal injection system is included as part of the chemical and volume control system (CVCS) charging subsystem and testing described in CESSAR-DC Section 14.2.12.1.8.

The staff has verified that CESSAR-DC Section 14.2.12.1.8 does include adequate testing of the RCP seal injection functions of the CVCS. This response is acceptable.

1.n.(15) Shield cooling systems

ABB-CE stated that shield cooling systems are not applicable because no such system is required for System 80+.

In the DSER, the staff asked ABB-CE to identify this exception to RG 1.68, Appendix A, in CESSAR-DC Section 14.2.7, "Conformance of Test Programs With Regulatory Guides." This was designated as DSER Confirmatory Item 14.2.12.2-3.

By letter dated April 21, 1993, ABB-CE submitted a draft of Section 14.2.7.1.11, which states "A shield cooling system is not a design requirement for the System 80+. This testing will not be performed as it is not applicable." ABB-CE incorporated this information into CESSAR-DC

in Amendment O. This is acceptable. On this basis, DSER Confirmatory Item 14.2.12.2-3 is resolved.

1.o.(1) Reactor component equipment operability at 100-percent load.

In Amendment J, ABB-CE revised CESSAR-DC Section 14.2.12.1.98, "Containment Polar Crane Test," to include performing an operational test of the polar crane at a 150-percent crane static load capacity test and a 100-percent crane rated load test. This is acceptable.

2.g Final calibration and proper operation of associated alarms and protective functions of source and intermediate range neutron flux monitors

In Amendment J, ABB-CE submitted a response to this item in CESSAR-DC Section 14.2.12.2.11, "Post-Core Ex-Core Nuclear Instrumentation Test." The staff reviewed the changes submitted for source and intermediate range neutron flux monitors in CESSAR-DC Section 14.2.12.2.11. These changes are acceptable.

2.c Final functional testing of the reactor protection system to demonstrate proper trip points, logic, and operability of scram breakers and valves; operability of manual scram functions

ABB-CE stated their intent to add a new CESSAR-DC Section 14.2.12.2.11, "Post-core Plant Protection (PPS) Test," to address this item. This was designated as DSER Open Item 14.2.12.2-2.

The staff verified that Amendment J to CESSAR-DC contained test abstract 14.2.12.2.11, "Post-core Ex-core Nuclear Instrumentation System Test." This test abstract verifies proper functional performance of the excore nuclear instrumentation system. However, the applicant did not submit adequate information concerning testing of the PPS in this test abstract. Subsequently, ABB-CE included in CESSAR-DC (Amendment Q) post-fuel load testing of the PPS in CESSAR-DC Section 14.2.12.29, "Post-core Instrument Correlation," including cross-reference acceptance criteria information to CESSAR-DC Sections 7.2 and 7.7. This is acceptable. On the basis, DSER Open Item 14.2.12.2-2 is resolved.

4.i Operability of the control rod withdrawal and insertion sequencer and control rod withdrawal inhibit or block functions over the reactor power level range during which such features must be operable

ABB-CE stated that these features are not operable at low power; therefore, no testing is performed.

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In the DSER, the staff asked ABB-CE to provide a test abstract to address this item which is scheduled to be conducted at the appropriate reactor power level range during which such features must be operable. This was designated as DSER Open Item 14.2.12.2-3.

ABB-CE responded by letter dated January 26, 1993, and the staff verified that control rod operability testing is specified in the CESSAR-DC test abstract 14.2.12.2.4. This information was incorporated into CESSAR-DC in Amendment O. This is acceptable. On this basis, DSER Open Item 14.2.12.2-3 is resolved.

Furthermore, the applicant was asked to identify this exception to RG 1.68, Appendix A, in CESSAR-DC Section 14.2.7, "Conformance of Test Programs With Regulatory Guides." This was designated as DSER Confirmatory Item 14.2.12.2-4.

ABB-CE responded by letter dated January 26, 1993, and the staff noted that ABB-CE intends to add Section 14.2.7.1.12, Reference Appendix A, Section 4.i, to state: "Demonstration of the operability of control rod withdraw and insertion sequence and control rod inhibit or block functions is performed during precritical functional testing. The reactor power level range during which such features must be operable is performed using simulated signals, as required." This response is acceptable. ABB-CE incorporated this information into CESSAR-DC in Amendment O. On this basis, DSER Confirmatory Item 14.2.12.2-4 is resolved.

4.t Natural circulation tests of the reactor coolant system

In CESSAR-DC Table 14.2-5, ABB-CE identifies the natural circulation test, but a test abstract for circulation was not submitted in CESSAR-DC Section 14.2.12. The applicant prepared a new CESSAR-DC Section 14.2.12.4.23, "Natural Circulation."

In accordance with the review guidance provided in RG 1.68, Revision 2, Appendix A, Regulatory Position 4.t, the staff concludes that the new test abstract in CESSAR-DC Section 14.2.12.4.23 is acceptable.

5.q Operation of failed fuel detection systems

ABB-CE stated that operation of the failed fuel detection system is included in CESSAR-DC Section 14.2.12.4.4, "Reactor Coolant and Secondary Chemistry and Radiochemistry Test."

The staff reviewed CESSAR-DC Section 14.2.12.4.4 and determined that the test abstract does include testing of the failed fuel detection system. This is acceptable.

5.w Demonstration that concrete temperatures surrounding hot penetrations do not exceed design limits with the minimum design capability of cooling system components available.

ABB-CE prepared a new CESSAR-DC Section 14.2.12.4.21, "Penetration Temperature Survey Test," to address this issue.

The staff reviewed CESSAR-DC Section 14.2.12.4.21 to verify the test abstract conformance to RG 1.68, Revision 2, Appendix A, Regulatory Position 5.W, and the staff concludes that the test abstract is acceptable.

5.x Auxiliary systems required to support operation of engineered safety features adequately perform under limiting accident conditions

ABB-CE stated their intent to prepare a new CESSAR-DC Section 14.2.12.4.22, "Ventilation Capability," to address this issue.

The staff found this intended change should address all auxiliary systems that are necessary to support operation of engineered safety features (ESF) actuation equipment, not just heating, ventilation, and air conditioning (HVAC) systems, (e.g., lube oil, bearing cooling, pneumatic devices). This was designated as DSER Open Item 14.2.12.2-4.

ABB-CE responded by letter dated January 26, 1993, and the staff noted that functional performance of all ESF systems and their auxiliary support systems is demonstrated during preoperational testing. The ventilation systems are expected to be affected by plant operating conditions and as such are the only ESF auxiliary support systems that will not be fully demonstrated by preoperational testing. This response is acceptable. ABB-CE incorporated this information CESSAR-DC in Amendment O. On this basis, DSER Open Item 14.2.12.2-4 is resolved.

5.c.c Gaseous and liquid radioactive waste processing, storage, and release systems

ABB-CE stated that operability of gaseous and liquid radioactive waste processing, storage, and release systems is verified in the preoperational tests described in CESSAR-DC Sections 14.2.12.1.114 and 14.2.12.1.116.

The staff determined that these sections include testing of gaseous and liquid radioactive waste systems during the preoperational test stage to satisfy most of the design requirements described in CESSAR-DC Sections 11.3 and 11.2, respectively. However, these tests are not performed during the power ascension test phase as is recom-

mended in RG 1.68 (Rev. 2), Appendix A, Regulatory Position 5.c.c. The applicant responded by incorporating tests for both these systems during the power ascension test phase in CESSAR-DC Amendment U. The staff reviewed the test abstracts to verify their conformance to RG 1.68, Revision 2, Appendix A, Regulatory Position 5.c.c. and finds them to be acceptable.

5.f.f Ventilation and air-conditioning systems

ABB-CE prepared a new CESSAR-DC Section 14.2.12.4.22, "Ventilation Capability," to address this issue.

The staff reviewed CESSAR-DC Section 14.2.12.4.22 to verify the test abstract conformance to RG 1.68, Revision 2, Appendix A, Regulatory Position 5.f.f, and determined that the abstract was acceptable.

14.2.12.3 Conformance With Other Regulatory Guides

Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants"

In the DSER, the staff asked ABB-CE to revise CESSAR-DC Section 14.2.12.4.7, "Shutdown From Outside the Control Room Test," to address the following items in accordance with RG 1.68.2:

- (1) Specify that plant systems are in the normal configuration with the turbine generator in operation.
- (2) Specify that the test is accomplished with the minimum shift crew complement.

ABB-CE revised CESSAR-DC Section 14.2.12.4.7 in a markup to specify that plant systems are in the normal configuration with the turbine generator in operation and that the test is accomplished with the minimum shift crew complement in accordance with RG 1.68.2.

The staff reviewed the proposed changes and determined they were acceptable subject to incorporation into the CESSAR-DC. This was designated as DSER Confirmatory Item 14.2.12.3-1.

The staff reviewed the changes that were incorporated into CESSAR-DC in Amendment J; they are acceptable. On this basis, DSER Confirmatory Item 14.2.12.3-1 is resolved.

Regulatory Guide 1.68.3, "Preoperational Testing of Instrument and Control Air Systems"

In the DSER, the staff asked ABB-CE to revise CESSAR-DC Section 14.2.12.1.88, "Compressed Air System Test," to address the following items in accordance with RG 1.68.3:

- (1) Determine that the total air demand at normal steady-state conditions, including leakage from the system, is in accordance with design requirements (Position C.5).
- (2) Demonstrate that the plant equipment designated to be supplied by the instrument air system is not degraded when supplied by the station air system which may have less-restrictive air quality requirements (Position C.9).

For Item 1 above, ABB-CE revised CESSAR-DC Section 14.2.12.1.88, "Compressed Air System Test," to reflect the guidance of RG 1.68.3, Regulatory Position C.5. The staff reviewed the changes to the compressed air system test in CESSAR-DC to verify test abstract conformance to RG 1.68.3, Regulatory Position C.5 (Amendment Q); they are acceptable. On this basis, DSER Confirmatory Item 14.2.12.3-2 is resolved.

For Item 2 above, ABB-CE stated that the instrument air system has no links to any other air system and, therefore, the ingress of air of a lesser quality is not possible and RG 1.68.3, Regulatory Position C.9, does not apply.

In the DSER, the staff asked ABB-CE to identify this exception to RG 1.68.3, Regulatory Position C.9, in CESSAR-DC Section 14.2.7, "Conformance of Test Programs With Regulatory Guides." This was designated as DSER Open item 14.2.12.3-1.

ABB-CE identified this exception to RG 1.68.3, Regulatory Position C.9, in a draft revision to CESSAR-DC Section 14.2.7, "Conformance of Test Programs With Regulatory Guides." In the revision, ABB-CE stated that "the CE System 80+ Instrument Air System has no interconnection to any other air compressed system; therefore, ingress of less quality air is not possible. Consequently, Position C.9 does not apply." The applicant submitted this information in CESSAR-DC (Amendment L). On this basis, DSER Open Item 14.2.12.3-1 is resolved.

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Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors"

The staff asked ABB-CE to revise CESSAR-DC Section 14.2.12.1.22 ("Safety Injection System Test"), CESSAR-DC Section 14.2.12.1.23 ("Safety Injection Tank Subsystem Test"), CESSAR-DC Section 14.2.12.1.41 ("Integrated Engineered Safety Feature/Loss of Power Test"), CESSAR-DC Section 14.2.12.1.61 ("Pre-core Safety Injection Check Valve Test"), or other test abstracts as appropriate to address the following items in accordance with RG 1.79:

- (1) Safety injection pump flow test under hot operating conditions (Position C.1.a.(2)).
- (2) Safety injection tank isolation valve test maximum differential pressure conditions (Position C.1.c.(2)). CESSAR-DC Section 14.2.7.2 addresses this testing but it is not specified in a test abstract.

ABB-CE revised CESSAR-DC Section 14.2.12.1.61 in Amendment Q to test the safety injection pump flow test under normal hot operating conditions to address RG 1.79, Regulatory Position C.1.a.(2). The staff reviewed the changes to CESSAR-DC Section 14.2.12.1.61 to verify the test abstract conformance to RG 1.79, Regulatory Position C.1.A(2), and the test abstract is acceptable.

ABB-CE revised CESSAR-DC Section 14.2.12.1.23 in Amendment Q to test the safety injection tank isolation valves under pressurized conditions. The test method includes observation of pressure indications, controls, and alarms needed to give the isolation valves the confirmatory open signal to address RG 1.79, Regulatory Position C.1.c.(2). The staff reviewed the changes to CESSAR-DC Section 14.2.12.1.23 to verify test abstract conformance to RG 1.79, Regulatory Position c.1.c.(2); the test abstract is acceptable.

Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against Accidental Chlorine Release"

The staff asked ABB-CE to revise CESSAR-DC Section 14.2.12.1.103, "Control Building Ventilation System Test," to address the concerns of RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against Accidental Chlorine Release."

ABB-CE revised CESSAR-DC Section 14.2.12.1.103 in Amendment Q to address the automatic transfer of the control building ventilation system to the emergency operations mode when a toxic chemical release is detected

in accordance with the recommendations of RG 1.95. The staff reviewed the changes to CESSAR-DC Section 14.2.12.1.103; the test abstract is acceptable.

Regulatory Guide 1.108, "Preoperational Testing of Diesel Generator Units as Onsite Power Systems at Nuclear Power Plants"

The staff asked ABB-CE to revise CESSAR-DC Section 14.2.12.1.94, "Diesel Generator Electrical System Test," or other test abstracts as appropriate to address the following items in accordance with RG 1.108:

- (1) Design-accident-loading-sequence to design-load requirements capability verification at full-load temperature conditions (Position C.2.a.(5)).
- (2) Consecutive start demonstration (Position C.2.a.(9)).

ABB-CE revised CESSAR-DC Section 14.2.12.1.94 in Amendment J to add test method 3.8 to address RG 1.108, Regulatory Position C.2.a.(5). The staff reviewed the changes to CESSAR-DC Section 14.2.12.1.94 to verify test abstract conformance to RG 1.108, Regulatory Position C.2.a.(5), and the staff considers the test abstract acceptable. It should be noted that RG 1.108 has been superseded by RG 1.9, Revision 3, and ABB-CE's test abstract is conformance with RG 1.9, Revision 3.

ABB-CE stated that the consecutive start demonstration of the diesel generators is performed in CESSAR-DC Section 14.2.12.1.94, "Diesel Generator Mechanical System Test." The staff verified that CESSAR-DC Section 14.2.12.1.93 (not CESSAR-DC Section 14.2.12.1.94 as stated) states that consecutive starts of the diesel generators will be performed. This is acceptable.

RG 1.108 has been withdrawn and is now superseded by RG 1.9, Revision 3, "Selection, Design and Qualification of Diesel Generator Units as Standby (Onsite) Electric Power Systems at Nuclear Power Plants," published July 1993. The contents of Section 14.2.12.1.94 are in accordance with the regulatory positions provided in RG 1.9, Revision 3; therefore, they are acceptable to the staff.

Regulatory Guide 1.139, "Guidance for Residual Heat Removal"

The staff asked ABB-CE to revise CESSAR-DC Section 14.2.12.1.21, "Shutdown Cooling System Test," to address the following items in accordance with RG 1.139:

- (1) RHR system isolation (Position C.2)

(2) RHR system pressure relief (Positions C.3)

ABB-CE revised CESSAR-DC Section 14.2.12.1.21 in Amendment Q to address RG 1.139, Regulatory Positions C.2 and C.3. The staff reviewed the test abstract and verified that the applicant addressed Position C.2 by providing test method 3.5 which requires testing of the residual heat removal (RHR) system isolation alarms and signals. This is acceptable.

ABB-CE should modify CESSAR-DC Section 14.2.12.1.21, test method 3.7 to state "verify the relieving capacity and setpoint of the Low Temperature Over-Pressure Protection (LTOP) relief valves" to comply with RG 1.139, Position C.3. The applicant responded by stating that the LTOP relief valve relieving capacity is conducted during bench testing before the valves are installed in the RHR system. Since the bench test verifies the relieving capacity, preoperational testing of the LTOP relief valve relieving capacity is not necessary. The staff stated that to conform to RG 1.139, the applicant should add this information to the prerequisite section to state that "the LTOP relief valves relieving capacity is verified by bench testing." The applicant will incorporate this information into the test abstract prerequisite section into the CESSAR-DC. This response is acceptable pending incorporation into the CESSAR-DC. This was identified as part of FSER Confirmatory Item 1.1-1 (see Chapter 1 of this report) in the advanced version of this report. ABB-CE has revised the CESSAR-DC Amendment V to include this information; this change is acceptable.

14.2.13 Security Considerations

The startup test program described in CESSAR-DC Section 14.2 includes security lighting system and security radio system tests. In CESSAR-DC Section 9.5.2, ABB-CE describes a security radio system for offsite communications. Security "lock-down" of the protected area and startup testing of the rest of the security system (i.e., intrusion detection system, alarm assessment system, access control system, etc.) were not addressed. In its December 17, 1991, response to RAI Q500.31, the applicant stated

Security 'lockdown' of the protected area and startup testing of the rest of the security system (i.e., intrusion detection system, alarm assessment system, access control system, etc.) is considered sensitive information which may be withheld from the public by the directive of 10 CFR 2.790(d). Full disclosure and description of these sensitive systems and their prerequisite testing will, however, be a part of the site security plan to be submitted by the utility.

The staff finds it acceptable that detailed description and NRC review of the rest of the security system and its test and acceptance criteria be deferred to the COL applicant's security plan submittal required by 10 CFR 50.34. This is COL Action Item 14.2.13-1. By letter dated January 26, 1993, ABB-CE noted in CESSAR-DC Section 14.2.7.5 that the COL applicant will provide site-specific security, contingency, and guard training plans. This is acceptable.

In RAI Q500.31, the staff noted that the security lighting system test described in CESSAR-DC Section 14.2.12.1.85 was incomplete in that it does not address testing on loss of normal power nor testing of its adequacy for support of closed-circuit television (CCTV) security functions. In its December 17, 1991 response, the applicant committed to amend subsections of the security lighting system test method to read:

- 3.3 Demonstrate that loss of normal power results in proper activation of the Security Lighting System for each affected room.
- 3.4 Demonstrate the Security Lighting System provides adequate illumination levels, including, but not limited to, those required to support plant Closed Circuit TV security functions.

These proposed changes did not adequately address the staff's concern regarding demonstrating that the system as installed has the capability described in the CESSAR-DC. In CESSAR-DC Section 9.5.3.1, ABB-CE states that the security lighting system will provide "illumination required to monitor isolation zones and all outdoor areas within the plant protected perimeter, under normal conditions as well as upon loss of all ac power." The proposed change addressed demonstrating lighting adequacy on loss of power "for each affected room" but not where "required to monitor isolation zones and all outdoor areas within the plant protected perimeter." This was designated as DSER Open Item 14.2.13-1. By letter dated April 21, 1993, ABB-CE modified CESSAR-DC Section 14.2.12.1.85 to require demonstration of the adequacy of illumination for CCTV security functions on loss of normal lighting power for monitored isolation zones and outdoor areas within the plant protected perimeter. ABB-CE incorporated this test abstract information into CESSAR-DC Amendment O. On this basis, DSER Open Item 14.2.13-1 is resolved.

Also in RAI Q500.31, the staff noted that the communications test described in CESSAR-DC Section 14.2.12.1.87 did not address verifying communication capability and noninterference with equipment from normal security patrol, alarm response, and interdiction paths within the plant. In its December 17, 1991, response, the applicant committed to amend subsections of the communications

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system test. This was designated as DSER Confirmatory Item 14.2.13-1. By letter dated April 4, 1993, the applicant added the following statement to CESSAR-DC Section 14.2.12.1.87:

- 3.4 Verify the Security Radio System functions properly at all locations throughout the plant.

This is acceptable. On this basis, DSER Confirmatory Item 14.2.13-1 is resolved.

14.2.14 Conclusion

On the basis of its review of the description of the ITP in CESSAR-DC, and the responses to the RAI items, and DSER open and confirmatory items, the staff concludes that the ITP is generally comprehensive and covers all areas of staff concern. The staff also conducted an in-depth review of system-specific testing requirements within each test abstract. The staff concludes that ABB-CE submitted a sufficient level of detail to adequately describe system-specific test prerequisites and acceptance criteria.

14.3 Certified Design Material

Introduction

The objective of this section of the chapter is to give the basis for the staff's approval of the certified design material (CDM) for the System 80+ design. This section is based on the staff's review of the ABB-CE document titled "System 80+ Certified Design Material" and ABB-CE's Standard Safety Analysis Report — Design Certification (CESSAR-DC) Section 14.3, "Certified Design Material." The requirement to submit this information as part of an application for design certification is in 10 CFR 52.47.

The ABB-CE document titled "System 80+ Certified Design Material," contains the information that will be certified by the rule that approves the System 80+ design. The CDM document has an introductory section, design descriptions and corresponding inspections, tests, analyses, and acceptance criteria (ITAACs) for the systems and structures of the design, additional design material applicable to multiple systems of the design, additional interface requirements, and site parameters for the System 80+ design. This information is also referred to as "Tier 1" information. In CESSAR-DC Section 14.3, "Certified Design Material," ABB-CE discusses the bases and methods that were used to develop the information for each of these CDM items for the System 80+. In this section, the staff documents its review of the System 80+ CDM,

as supported by the design information in the System 80+ CESSAR-DC.

The information in the CDM and CESSAR-DC Section 14.3 comes from the detailed information in the CESSAR-DC. Further, the purpose of the ITAACs, which are part of the CDM, is to verify that a facility that references the design certification has been built and will operate in accordance with the design certification and the applicable regulations. Consequently, there is no design information presented in the CDM or CESSAR-DC Section 14.3 that is not also in the appropriate sections of the CESSAR-DC. Therefore, the staff did not base its safety evaluations of the design on the information in the CDM.

Basis for Approval of the CDM

The evolutionary reactor designs were the first designs to develop CDM for design certification. ABB-CE closely followed the development of the initial CDM for the lead evolutionary design. Although the staff was able to make its safety determinations for the design based on its review of the information in the CESSAR-DC, the System 80+ CDM was among the first design material that had been developed for certification. Therefore, there was only the precedent established by the lead design for ABB-CE to follow to develop information for the CDM. Furthermore, the staff had no formal regulatory guidance on which to base its review, relying in many cases on the lessons learned from the review of the CDM for the lead design. The staff sought and received Commission guidance on several key issues associated with the design certification reviews in the staff requirements memorandum (SRM) dated February 15, 1991, relating to SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52." These issues included the development of regulatory guidance, the role of ITAACs, the level of design detail needed for design certification, issue finality, the two-tiered approach to the design certification rule structure, and flexibility in the design change process. In its review, the staff assessed the CDM in the context of Part 52 requirements and the Commission guidance contained in that SRM.

ABB-CE submitted the CDM to the staff in stages, so that lessons learned at each stage could be incorporated into later submittals. The staff reviewed the material and sent comments on the CDM to ABB-CE at each stage. ABB-CE and the staff closely followed the developments and resolutions of CDM issues for the lead design. The staff informed the Commission of the development of the CDM for the lead design in many SECY papers issued in 1991 and 1992 (SECYs 91-178, 91-210, 92-053, 92-196,

92-214, 92-287 and 287A, 92-299, and 92-327). These papers are listed as references in Table 1.2 of this report.

In CESSAR-DC Section 14.3, ABB-CE described the process it used in the development of the CDM, based on the design presented in the appropriate sections of the CESSAR-DC. ABB-CE provided the CDM based on the structures and systems of the System 80+ design rather than based on the format of the CESSAR-DC. In addition, ABB-CE adopted a graded approach to the level of design detail for the information in the CDM, based on the safety significance of particular structures, systems, and components (SSCs). ABB-CE applied various selection criteria to the information in the CESSAR-DC to determine the level of design information for a given structure or system in the CDM. The results of this process were illustrated with cross-references from the CESSAR-DC information to the CDM for important parameters that were selected for treatment in the CDM. Although many issues and analyses could have been cross-referenced, the listings in Section 14.3 were developed only for selected integrated plant safety analyses for the System 80+ design. ABB-CE provided additional cross-references of key insights and assumptions from probabilistic risk assessments (PRAs) and analyses for severe accidents which are in CESSAR-DC Section 19.15. ABB-CE provided more detailed cross-references to the CDM for these analyses in a letter dated June 10, 1994.

The staff also utilized a graded approach to the level of detail in its review of the CDM based on the safety significance of the SSCs. Thus, consistent with the guidance of 10 CFR Part 52 and the SRM related to SECY-90-377, the staff recognized that although many aspects of the design were important to safety, the level of design detail in the CDM and verification of the key design features and performance characteristics should be commensurate with the significance of the safety functions to be performed.

The intent of the CDM is to ensure that the key characteristics and performance requirements of safety-significant SSCs are implemented in an as-built facility referencing the certified design. Although all these aspects of the design are described in the CDM, not all can be verified by the ITAACs because Part 52 requires that the ITAACs be satisfied before fuel loading. The initial test program (ITP) serves to verify the remaining aspects of the design after fuel load, but before operation. Examples of these requirements are the post-fuel load startup and power ascension test program verification of fuel, control rod, and core characteristics, as well as system and integrated plant operating characteristics. The treatment of these issues will be similar to their treatment at facilities licensed under 10 CFR Part 50, in that verification of the satisfacto-

ry completion of these requirements will be a condition of the license.

The staff recognized that other programs also ensure the continued safe operation of a facility after fuel load. For example, the continued operability of a facility after the ITAACs are satisfied is ensured through the technical specifications, as well as various programs, such as the maintenance program, quality assurance program, and the in-service inspection and testing program. The operator ensures the facility is operated as designed, through the use of appropriate plant operating and emergency procedures. Additionally, a utility referencing the design is required by 10 CFR Part 50 (Appendix B) to have a quality assurance (QA) program that ensures these SSCs are appropriately designed and procured, and perform satisfactorily in service.

These considerations became an overall framework for ABB-CE's development of the CDM and the staff's review. The staff utilized many sources of information to determine the safety significance of SSCs in the CDM. Among these sources were the SRP, applicable rules and regulations, general design criteria (GDC), regulatory guides (RGs), unresolved safety issues (USIs) and generic safety issues (GSIs), NRC generic correspondence, operating experience, NRC inspection programs, facility testing programs, PRA, and insights from System 80+ safety analyses and severe-accident analyses. For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs."

Nonetheless, because the evolutionary designs were the first designs to develop CDM for design certifications, because there was only the precedent of the lead evolutionary design for ABB-CE to follow and no formal review guidance for the staff to base its review on, and because of the iterative nature of the ITAAC development process, considerable judgment was inherent in the approval of the final material for the CDM.

Background

Part 52 of Title 10 of the Code of Federal Regulations was issued on April 18, 1989. The concept in Part 52 of certifying a design and specifying the required ITAACs in a rule preceding construction of a facility had not been attempted before. Consequently, the Nuclear Management and Resources Council (NUMARC) held extensive discus-

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sions with the staff in 1990 and 1991 on the CDM, especially the form and content of the ITAACs. After several senior management meetings in mid-1991, an evolutionary design emerged as the lead plant in the development of the first CDM for a standardized design.

The development of the CDM was an iterative process. ABB-CE submitted the CDM to the staff for review in increments, so that lessons learned at each stage could be incorporated into later submittals. The staff reviewed the material as it was submitted and commented on the CDM to ABB-CE at each stage. Senior managers from ABB-CE and the staff met periodically to resolve difficult policy and technical issues associated with the development and review of the material. ABB-CE and the staff closely followed the developments and resolutions of CDM issues for the lead design.

ABB-CE first submitted the CDM, consisting of a set of ten "pilot" ITAACs on April 30, 1992. The staff commented on this submittal in a letter dated May 21, 1992, and after ABB-CE met with the staff to discuss the comments, general agreement was reached on the pilot ITAACs in May 1992.

ABB-CE revised the pilot ITAACs on the basis of the staff's comments and lessons learned from the lead evolutionary design, and resubmitted them in a letter dated August 10, 1992. The staff commented to ABB-CE on the revised CDM in a letter dated October 19, 1992, as well as during several senior management meetings.

ABB-CE submitted the initial complete CDM for the System 80+ in letters dated January 28 and February 1, 1993. However, in February 1993, a group comprised of representatives from various vendors, utilities, and industry groups submitted comments on the initial CDM to ABB-CE. ABB-CE revised the initial CDM to incorporate the comments submitted by this review group, and sent a set of "prototype" CDM for review by the NRC in March 1993. Subsequently, a team of NRC reviewers met with ABB-CE in March 1993 to review the prototype CDM.

ABB-CE submitted a revised, complete CDM that incorporated all previous lessons learned and review comments, in incremental submittals in April, May, and June of 1993. In May 1993, the staff formed several task groups to perform a multidisciplinary review of the CDM. The task groups sent extensive comments on the submittals to ABB-CE in a letter dated September 2, 1993. The task groups met with ABB-CE on October 4 through 6, 1993, to discuss the comments, and the disposition of all of the comments was documented in a meeting summary dated October 15, 1993.

ABB-CE submitted another revised, complete CDM that incorporated the comments from the staff's task group review in a document titled "System 80+ Certified Design Material," and dated December 31, 1993. Changes to the detailed CDM supporting information in the CESSAR-DC were contained in supporting amendments to the CESSAR-DC.

The staff performed an independent review of the CDM and the CESSAR-DC in January 1994 to ensure that the design information in the documents was consistent. The staff performed this review because the information in the CDM is certified by the System 80+ design certification rulemaking, is incorporated into the rule and is therefore difficult to change, and consequently must reflect the design accurately.

The independent review group completed its review in March 1994, and its comments were provided to ABB-CE after review by the staff. ABB-CE provided satisfactory resolutions to all the comments in a CDM submitted with CESSAR-DC Amendment W. In addition to resolving the comments of the independent review group, this revision of the CDM also included resolutions to comments by the ACRS and the staff.

Format of the CDM

ABB-CE developed the design certification information to comply with the requirements of 10 CFR 52.47, including design descriptions and corresponding ITAACs, interface requirements for the design, and bounding parameters for siting of the standard design. The CDM is the portion of the design information that is certified by the rule certifying the design, and will be incorporated into the rule as the Tier 1 part of the design control document (DCD). The DCD is the master document that contains the information that must be conformed with by an applicant who references the rule. The format for this material is listed below, and is discussed in more detail in subsequent sections of this chapter.

- (1) Introduction - Definitions of terms used in the CDM, and a listing of general provisions that are applicable to all CDM entries.
- (2) System Design Descriptions and ITAACs - System design descriptions and ITAACs are provided for:
(a) structures and systems that are fully within the scope of the System 80+ design certification and
(b) the in-scope portions of those systems that are only partially within the scope of the System 80+ design certification. The system design descriptions are accompanied by the appropriate ITAACs.

- (3) Additional Certified Design Material - Design descriptions and their related ITAACs for design and construction activities that are applicable to more than one system of the design. This additional material was provided because in selected areas of the design, ABB-CE did not submit sufficient design detail in the CESSAR-DC. ABB-CE did not submit complete design information in these areas because the information was dependent on as-built or as-procured information. For these areas, ABB-CE provided the design related processes in the CDM and in the CESSAR-DC, with appropriate codes and standards, that a COL applicant or licensee would adhere to, in order to complete the design.
- (4) Interface Requirements - Requirements that must be complied with by the site-specific portions of a facility that are not within the scope of the certified design. This section also identifies the scope of the design to be certified. Safety-significant interface requirements are defined for: (a) systems that are entirely outside the scope of the design, and (b) the out-of-scope portions of those systems that are only partially within the scope of the System 80+ design.
- (5) Site Parameters - Bounding parameters of the design to be used in the selection of a suitable site for a facility referencing the System 80+ certified design. The design was evaluated in terms of these parameters. A suitable site must be demonstrated to be within the bounding parameters and characteristics, and a facility must be constructed at the site in accordance with their use in the approved design. If a site cannot conform to them, an exemption must be requested, and the facility must be reevaluated in terms of these parameters for the actual selected site.

14.3.1 Introduction

14.3.1.1 Definitions

This section of the CDM provides terms used in the CDM that could be subject to various interpretations. The intent of the terms used in the CDM was to be consistent and as closely aligned as possible with the terminology in the CESSAR-DC, in common industry use, in industry codes and standards, and in NRC rules, regulations, and guidance. Thus, should questions on terminology arise, these references would aid in understanding the intent of the information in the CDM.

14.3.1.2 General Provisions

This section of the CDM provides general provisions that are applicable to the design descriptions, figures, and the ITAACs.

14.3.1.2.1 Verifications for Basic Configuration for Structures and Systems

This section of the CDM includes provisions related to the verification of the ITAACs for basic configuration for systems and structures of the design. These ITAACs are contained in the buildings and many of the systems described in Section 2.0 of the CDM. The verification consists of an inspection of the system functional arrangement in its final as-built condition at the plant site, and includes the elements of the design descriptions and the system figures in the CDM. This functional arrangement inspection verifies, using as-built system drawings, design documentation, and in situ plant walkdowns, that the as-built facility is in conformance with the certified design and applicable regulations.

Several other aspects of the design were considered to have significance to the performance of safety functions of SSCs of a facility. The basis for selecting these aspects included its importance to safety as well as its past experience with construction and operating problems. Thus, specific inspections for these aspects are part of the basic configuration ITAACs for systems and structures. The other inspections to be conducted to satisfy these ITAACs include, and are limited to, verification of the following:

- (1) Quality of pressure boundary welds for ASME Code Class 1, 2, and 3 components and systems described in the design descriptions and figures. Detailed supporting information for verification of welding requirements in accordance with ASME Code requirements is contained in CESSAR-DC Chapter 3.
- (2) Dynamic qualification (e.g., seismic, LOCA, and safety/relief valve discharge loads) of seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) described in the design descriptions and figures. Detailed supporting information for dynamic qualification requirements, including qualification records, is contained in CESSAR-DC Chapter 3.
- (3) Environmental qualification of Class 1E electrical equipment described in the design descriptions and figures. Detailed supporting information for environmental qualification requirements is contained in CESSAR-DC Chapter 3.

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- (4) Design qualification of motor-operated valves (MOV) described in the design descriptions and figures. Detailed supporting information for design qualification of MOVs is contained in CESSAR-DC Chapter 3.

14.3.1.2.2 Treatment of Individual Items

A licensee referencing the standard design is not prohibited from utilizing an item not described in the CDM. However, the as-built facility must be consistent with the rule approving the design, including both tiers of information. The change processes for the certified design are described in the design certification rule for the System 80+.

The term "operate" as utilized in the CDM is intended to refer to the actuation and running of equipment. This is not meant to include the term "operable" in the context of the ongoing reliability and availability of equipment. In developing the ITAACs, the staff recognized that other programs ensure the continued safe operation of a facility after fuel load. For example, the continued operability of a facility after the ITAACs are satisfied is ensured through the technical specifications, startup and power ascension test programs, as well as through various programs such as the maintenance program, quality assurance program, and the in-service inspection and in-service testing program. Also, the operator ensures the facility is operated as designed, through the use of appropriate plant operating and emergency procedures.

The term "exists," when used in the acceptance criteria, means that the item is present and conforms to the design description. Detailed supporting information on what must be present to conclude that an item "exists" and conforms to the design description is contained in the appropriate sections of the CESSAR-DC.

14.3.1.2.3 Implementation of ITAACs

ABB-CE utilized a three column format for the ITAACs. The design commitments in the first column are derived from the design information in the design descriptions. The inspections, tests, and analyses in the middle column denote the intended means of verifying the design commitment. The acceptance criteria in the third column give the criteria used to determine whether the design commitment is met.

The licensee is required by 10 CFR Part 52 to perform the required inspections, tests, and analyses for the design, and to certify to the NRC that the acceptance criteria have been met. A licensee may utilize the efforts of subordinate vendors, contractors, or consultants. However, the licensee referencing the certified design retains responsibility

for ensuring that the ITAACs are met. Additionally, the ITAACs can be satisfied using other programs, such as the pre-operational testing portion of the initial test program (ITP) required by CDM Section 3.5, or the QA program required by 10 CFR Part 50, Appendix B.

The ITAACs may be satisfied at any time preceding fuel load, including before issuance of a combined license. However, the primary intent of the ITAACs is to verify that the as-built plant on the final site has been constructed and will perform in accordance with the design certification and applicable regulations.

14.3.1.2.4 Discussion of Matters Related to Operations

Descriptions in the CDM may refer to matters of operation, such as normal valve or breaker alignment during normal operational modes. These descriptions are not intended to require operators to take any particular action. The operational matters referred to in the CDM are governed by existing programs to ensure the ongoing safe operation of a facility, such as plant operating and emergency procedures.

14.3.1.2.5 Interpretation of Figures

The design descriptions include the figures in the CDM, where the figures are provided. They are intended to depict the functional arrangement of the significant SSCs of the System 80+ design. An as-built facility referencing the certified design must be consistent with the performance characteristics and functions described in the design descriptions and figures. Any changes to the detailed information in the CESSAR-DC must be made in accordance with the "50.59-like" change process in the design certification rule for the System 80+, which allows the COL applicant or licensee to make design changes, provided the changes do not impact the information in the CDM.

14.3.1.2.6 Rated Reactor Core Thermal Power

The rated reactor core thermal power for the System 80+ is 3914 MW(th).

14.3.1.3 Conclusions

As discussed above, the staff reviewed the definitions and general provisions that are contained in CDM Section 1.0, and the supporting information contained in CESSAR-DC Section 14.3.1, in accordance with the requirements in Part 52 and the guidance in SRMs related to design certification applications provided by the Commission. Based on this, the staff concludes that the definitions and general provisions

sions in the CDM are appropriate to support the design descriptions and ITAACs, and are acceptable.

14.3.2 Certified Design Material for Structures and Systems

ABB-CE developed design descriptions and ITAACs for the structures and systems of the System 80+ design, and these are in CDM Section 2.0. General provisions that apply to most of the structures and systems are in CDM Section 1.2. Additional CDM material for design issues that apply to many of these structures and systems are in CDM Section 3.0. Safety-significant interface requirements for these systems are in the system design descriptions for the in-scope portions of the systems. The safety-significant interface requirements for the out-of-scope portions of the systems of the design are contained in CDM Section 4.0.

14.3.2.1 Design Descriptions

The design descriptions address the most safety-significant aspects of each of the systems of the design, and were derived from the detailed design information contained in the CESSAR-DC. The design descriptions include the figures associated with the systems. ABB-CE's selection criteria and methodology for the system design descriptions are specified in CESSAR-DC Section 14.3.2.1.

The Tier 1 design descriptions will serve as commitments for the lifetime of a facility. Once completion of ITAACs and the supporting design information demonstrate that the facility has been properly constructed, it then becomes the function of existing programs such as the technical specifications programs, the in-service inspection program, the in-service testing program, the quality assurance program, and the maintenance program, to demonstrate that the facility continues to operate in accordance with the certified design and the license. Nevertheless, the Tier 1 design descriptions will remain in effect throughout the plant life to assure that the plant does not deviate from the certified design. In general, a COL applicant or licensee may change the information in the CESSAR-DC in accordance with the "50.59-like" change process described in the rule certifying the design, provided that the change does not impact the information in the design descriptions.

ABB-CE submitted the selection criteria for information in the design descriptions in CESSAR-DC Section 14.3.2.1. Essentially, ABB-CE put the top-level design features and performance standards that were most significant to safety in the design descriptions. The criteria ABB-CE utilized in determining the safety significance of SSCs in the design descriptions included the NRC's regulations, whether or

not the information pertained to safety-related SSCs, the importance in the SRP, the relative importance based on PRA or severe accident analysis, operating experience, or the technical specifications. ABB-CE also included other SSCs based on their importance to safety. Non-safety aspects of SSCs were generally not discussed in the design descriptions. Thus, the amount of information provided in the CDM for a given system was based on the safety significance of the system.

ABB-CE provided additional certified design material applicable to the systems of the design in CDM Section 3.0. The design descriptions in CDM Section 3.0 describe the scope and applicability of the additional CDM to the systems of the design in CDM Section 2.0. Amplifying information on CDM Section 3.0 is in CESSAR-DC Section 14.3.3, and the staff's review of CDM Section 3.0 is in Section 14.3.3 of this chapter.

The CDM utilizes a system-based structure which is different from the structure of the CESSAR-DC. Consequently, developing the CDM design description entries for any one system was based on a review of the multiple CESSAR-DC chapters having technical information related to that system. The level of design detail in any single system is proportional to the safety significance of the system. ABB-CE discusses this in CESSAR-DC Section 14.3.2.1, and illustrates the many design aspects of the systems, such as system classification, modes of operation, independence, and instrumentation that were considered, and were derived from the information in several chapters of the CESSAR-DC.

The staff was particularly interested in ensuring that the assumptions and insights from key safety and integrated plant safety analyses in the CESSAR-DC, where plant performance was dependent on contributions from multiple systems of the design, were adequately considered in the CDM. Addressing these assumptions and insights in the CDM ensures that the integrity of the fundamental analyses for the design are preserved in an as-built facility referencing the certified design. These analyses included flooding analyses, overpressure protection, containment analyses, core cooling analyses, fire protection, transient analyses and anticipated transients without scram (ATWS), and radiological analyses.

ABB-CE submitted information about these analyses in the CESSAR-DC, and documented the important design information and parameters from the various chapters of the CESSAR-DC that are addressed in the CDM in CDM Tables 14.3-1 through 14.3-7. These tables include PRA and severe accident analyses. ABB-CE also provided cross-references in CESSAR-DC Section 19.15 showing how key insights and assumptions from PRA and severe

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accident analyses are addressed in the CDM. ABB-CE provided more detailed cross-references to the CDM for these analyses in a letter dated June 10, 1994. A COL applicant or licensee proposing to change design information in the CESSAR-DC that pertained to these analyses via the "50.59-like" change process can use these cross-references when considering whether the proposed change impacts the treatment of these parameters in the CDM.

14.3.2.2 ITAACs

The purpose of the ITAACs is to verify that an as-built facility conforms to the approved plant design and applicable regulations. When coupled in a COL with the ITAACs for site-specific portions of the design, they constitute the verification activities for a facility that must be successfully completed before fuel load. If the licensee demonstrates that the ITAACs are complied with and the staff agrees that this is so, then the licensee will be permitted to load fuel.

The scope of the ITAACs is consistent with the SSCs that are in the design descriptions. In general, each system has one ITAAC or more that verifies the information in the design descriptions. This is not true in all cases. Reasons for not requiring an ITAACs verification for a Tier 1 design commitment include: (1) the information is only included for context, (2) fulfillment of other ITAACs are sufficient to show verification of the design commitment, (3) a single ITAACs can verify more than one design commitment, or (4) the item can only be verified after fuel loading. For the last item, the staff reviewed the power ascension testing program described in CESSAR-DC Chapter 14 to ensure that all important design features and commitments that could not be verified before fuel load were addressed where appropriate.

The staff reviewed the system ITAACs to ensure that the verifications were consistent with the safety significance of the key design characteristics and performance requirements of the SSC verified by that ITAACs. The certified design descriptions for an SSC contain the significant functions and bases for that SSC. Therefore, the ITAACs have been reviewed to ensure they are necessary and sufficient to provide the NRC with reasonable assurance that the facility should be authorized to load fuel. As a result, the ITAACs verify the significant design features from the design descriptions and the applicable requirements that are necessary and sufficient to authorize fuel loading and subsequent operation.

The staff and industry reached agreement on a three-column format for ITAACs, as discussed below.

Column 1 - Design Commitment

This column contains the text for the specific design commitment that was extracted from the design descriptions discussed above. Any differences in text were minimized, unless intentional. Differences in text were generally intended to better conform the commitments in the design description with the ITAAC format.

Column 2 - Inspections, Tests, and Analyses

This column contains the specific method to be used by the licensee to demonstrate that the design commitment in Column 1 has been met. The method is either by inspection, test, or analysis, or by some combination of inspections, tests, or analyses. Differences in text were generally intended to better conform the commitments in the design description with the format of the ITAACs.

The CESSAR-DC contains detailed supporting information for the CDM about various inspections, tests, and analyses that can, and should be, used to verify the Tier 1 design information and satisfy the acceptance criteria. If questions on interpretation should arise, the material in the CESSAR-DC provides the background material and context for the CDM. The CESSAR-DC contains information reviewed by the staff which was the basis for the staff's safety determination for the design. Therefore, the information in the CESSAR-DC provides an acceptable means of satisfying an ITAACs.

Inspections are defined in CDM Section 1.1, and include visual and physical observations, walkdowns, or record reviews. The inspections required for the "Basic Configuration Walkdown" ITAACs invoke the general provisions contained in CDM Section 1.2 for as-built structures and systems.

Tests are defined in CDM Section 1.1, and mean the actuation, operation, or establishment of specified conditions to evaluate the performance or integrity of the as-built SSCs. This includes functional and hydrostatic tests for the systems. The term "as-built" is intended to mean testing in the final as-installed condition at a facility. The term "type tests" is used in this column to mean manufacturer's tests or other tests that are not necessarily intended to be in the final as-installed condition. The results of pre-operational tests can be used to satisfy an ITAACs. In its review, the staff did not rely on the pre-operational tests described in CESSAR-DC Section 14.2 or RG 1.68 to substitute for ITAACs. Where testing is specified, appropriate conditions for the test should be established in accordance with the ITP described in CDM Section 2.11, CESSAR-DC Section 14.2, and RG 1.68. Conversion of the test results from the test conditions to

the design conditions may be required to satisfy the ITAACs.

During its review, the staff emphasized in-situ testing, where possible, of the as-built facility as the preferred means to satisfy the ITAACs. Also, the staff recognized that the results and documentation from facility programs such as the quality assurance program or the ITP may be used to satisfy an ITAACs.

Analyses are defined in CDM Section 1.1, and may refer to detailed supporting information in the CESSAR-DC, simple calculations, or comparisons with operating experience or design of similar SSCs. For example, detailed analysis methods of dynamic and environmental qualification supporting CDM Section 1.2 are in CESSAR-DC Chapter 3, and detailed piping design information supporting CDM Section 3.1, are also in CESSAR-DC Chapter 3.

Column 3 - Acceptance Criteria

This column contains the specific acceptance criteria for the inspections, tests, or analyses described in Column 2 which, if met, demonstrate that the design commitment in Column 1 has been met.

In general, the acceptance criteria were developed to be objective and unambiguous. In some cases, the acceptance criteria were more general because the detailed supporting information in the CESSAR-DC did not lend itself to concise verification. For example, the acceptance criteria for the design integrity of piping and structures is that a report "exists" that concludes the design commitments are met. In these cases, the CESSAR-DC provides the detailed supporting information on multiple interdependent parameters that must be provided in order to demonstrate that a satisfactory report exists.

Numeric performance values for SSCs were specified as ITAACs acceptance criteria when the design commitment so lent itself, or when failure to meet the stated acceptance criterion would clearly indicate a failure to properly implement the design. The staff did not require that numeric performance values be specified in the design description unless there was a specific reason to include them (e.g., important to be maintained for the life of the facility).

ITAACs Implementation

The ITAACs may be satisfied at any time before fuel load, as well as any time before the issuance of a combined license. However, the primary intent of the ITAACs is to verify the as-built facility on the final site has been

constructed and will operate in accordance with the design certification.

The implementation of a construction verification program, including ITAACs, and other licensee programs, is the responsibility of the licensee. The successful completion of the ITAACs in the combined license will constitute the basis for the NRC's determination to allow fuel loading for the facility.

The licensee will periodically certify to the NRC that the inspections, tests, and analyses have been performed, and that the acceptance criteria have been met. These notifications should document the basis for the successful completion of the ITAACs. In accordance with 10 CFR 52.99, the staff will ensure that the required inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. At appropriate intervals, the NRC will publish in the Federal Register, notices of the successful completion of the inspections, tests, and analyses.

14.3.2.3 Staff Review Approach

ABB-CE developed the CDM based on the systems of the design rather than on the format of the CESSAR-DC and SRP. In order to ensure that the safety-significant design information in the CESSAR-DC was adequately reflected in the CDM, the staff adopted a multidisciplinary review approach, rather than the more traditional review approach based on the individual chapters of the CESSAR-DC. The staff formed several task groups comprised of various disciplines to ensure that the CDM would provide reasonable assurance that a facility would be built and operated in accordance with the design certification and applicable regulations.

The task groups were composed of various representatives from the technical branches of the staff, depending on the primary area of review by the task group. The task groups were formed according to the following discipline areas: plant systems, reactor systems, electrical systems, human factors, radiation protection, structural systems, and instrumentation and controls.

The task groups had primary review responsibilities for systems that were predominantly in their discipline area, and secondary review responsibilities for systems in other appropriate disciplines based on the safety significance of the issues. Thus, the groups had overlapping review responsibilities. All information in the CDM was reviewed by one or more task groups. For example, the safety injection system was reviewed primarily by the Reactor Systems Task Group, and that task group received technical input and comments from the Instrumentation and

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Controls Task Group. Specialists were designated to provide input to the task groups for selected design issues. Examples of these issues included severe accident issues; testing issues and the ITP; treatment of alarms, displays, and controls; insights from PRA; and functionality of MOVs.

The task groups with primary review responsibility for particular systems maintained overall responsibility for the reviews of those systems. Overall continuity and consistency of the reviews was maintained through frequent meetings with all task groups and with the projects branches. Significant policy and technical issues, or issues of concern to multiple task groups, were identified for discussion at periodic senior management meetings between the staff and ABB-CE.

The staff developed preliminary draft guidance for use in the reviews of the CDM, and incorporated lessons learned during the course of the reviews into the draft guidance. The draft guidance contained checklists for use in the reviews. The applicability of the issues identified in the checklists to the systems was based on the safety significance of the specific SSCs. The draft guidance also contained standard ITAACs entries that were used to verify selected issues in the appropriate systems of the design. Examples of these standard ITAACs entries are those for the basic configuration of systems, verification of control room and remote shutdown features, and electrical independence. The issues in the checklist and the use of the standard ITAACs entries are discussed in the following sections of this report.

The task groups utilized multiple sources of information to determine the safety significance of SSCs in the CDM. These sources included the SRP, applicable rules and regulations, GDC, RGs, USIs and GSIs, NRC generic correspondence, operating experience, NRC inspection programs, facility testing programs, PRA, and insights from System 80+ safety and severe accident analyses.

For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," and the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." These issues included staff positions that deviated from or were not embodied in current regulations applicable to the System 80+ design. The staff's positions and design-specific requirements are addressed where appropriate in this section of this report, and in the System 80+ design certification rule as "applicable regulations."

The staff determined that selected material in the CESSAR-DC that supports the CDM, if considered for a change by an applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question and, therefore, would require NRC review and approval before implementation of the change. The material supporting the CDM is discussed where appropriate in this section, in the applicable chapters of this report, and in the System 80+ design certification rule.

The task groups utilized a graded approach to the level of detail in reviewing the CDM based on the safety significance of the SSCs. Thus, consistent with the guidance of Part 52 and the SRM related to SECY-90-377, the staff recognized that although many aspects of the design were important to safety, the level of design detail in the CDM and verification of the key features and performance characteristics should be commensurate with the significance of the safety functions to be performed. In addition, the CESSAR-DC was reviewed to ensure that the information was consistent with the design description and that the information supporting the Tier 1 material was comprehensive and technically adequate. Thus, the individual task groups reviewed the CDM based on the safety significance of the material, as discussed in the following paragraphs.

14.3.2.3.1 Plant Systems Task Group Review

The Plant Systems Task Group had primary review responsibility for most of the fluid systems in CDM Section 2.0 that were not part of the core reactor systems. The scope of the plant systems review included new and spent fuel handling systems, power generation systems, air systems, cooling water systems, and heating, ventilation and air conditioning systems. The group also reviewed selected interface requirements within those systems. The group reviewed issues that affect multiple SSCs, such as equipment qualification and protection from fires, floods and tornado missiles, and had secondary review responsibilities for radioactive waste systems, most of the fluid systems, and the structures of the design.

The task group primarily utilized the SRP in its review of the CDM to determine the safety significance of SSCs. Other sources included applicable rules and regulations, GDC, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from System 80+ safety and severe accident analyses, and operating experience. The task group also used the draft review guidance for the design control document as an aid in its review of the systems. For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory

Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." The task group reviewed the Tier 1 submittals (including the design description, figures, and ITAACs) of the design using the guidelines provided in the draft review guidance for the CDM as an aid for establishing consistency and completeness.

The task group reviewed the CDM for treatment of design information proportional to the safety significance of the SSC for that system. Many items were judged to be important to safety, and were thus included in the CDM. The following issues were identified to ensure comprehensive and consistent treatment in the CDM based on the safety significance of the system being reviewed:

- system purpose and functions
- location of system
- key design features of the system
- seismic and ASME Code classifications
- system operation in various modes
- controls, alarms, and displays
- logic
- interlocks
- Class 1E electrical power sources and divisions
- equipment to be qualified for harsh environments
- interface requirements
- numeric performance values
- accuracy and quality of figures

Additionally, standard ITAACs entries were utilized to verify selected issues, where appropriate. Examples of these included basic configuration, physical separation, and divisional power supplies. In particular, the general provision for environmental qualification aspects of SSCs invoked by the basic configuration ITAACs was reviewed to ensure appropriate treatment in the CDM.

Environmental qualification (EQ) of safe-shutdown equipment is verified as part of the basic configuration ITAACs for safety-related systems. EQ treatment in the ITAACs is discussed under "General Provisions" in the CDM. Verification includes type tests or a combination of type tests and analyses of Class 1E electrical equipment identified in the "Design Description" section or accompanying figures to show that the equipment can withstand the conditions associated with a design-basis accident without loss of safety function for the time that the function is needed.

The task group reviewed integrated plant safety analyses such as fires, floods and missile protection to ensure they were adequately addressed in the CDM. The insights from these analyses that were addressed in the CDM are

in CESSAR-DC Section 14.3. The issues of floods, fires, missiles, pipe failures, and environmental protection are verified by the ITAACs on a system-specific basis, rather than generically. Divisional separation (both physical and electrical) is the primary means of ensuring protection of safety-related equipment from these events. Divisional separation is verified as part of both individual system ITAACs and building ITAACs. Physical and electrical separation is verified in each safety-related system ITAACs and divisional barriers are verified in the nuclear island structures ITAACs.

The design features in the CDM were selected to ensure that the integrity of the analyses would be preserved in an as-built facility. For example, 3-hour-rated fire boundaries and divisional separation were shown in the building figures. Also, flooding features such as structure elevations were specified in the site parameters, flood doors and flood barriers were shown on the building figures, and elevations were shown on the buildings to verify that the approximate physical location of components and relative elevations of buildings minimized the effects of flooding. An as-built reconciliation report for fires to ensure consistency with the CESSAR-DC analyses is required by the fire protection system ITAAC.

Other specific issues that were addressed include heat removal capabilities for design-basis accidents and tornado and missile protection. Heat removal capabilities were verified through heat removal requirements for core cooling system heat exchangers and interface requirements for site-specific systems. Tornado and missile protection was provided by inlet and outlet dampers in ventilation systems, and through the structural design of buildings.

The task group received information from other task groups such as the structural, electrical, and I&C task groups. The task group also reviewed the ITAACs for consistency with the initial test program described in CESSAR-DC Chapter 14. In addition, specialists provided key insights and assumptions from PRA and severe accident analyses, as well as inputs for issues such as treatment of alarms, displays and controls, and functionality of MOVs. A cross-reference from the CESSAR-DC to the CDM providing these key insights and assumptions is in CESSAR-DC Section 19.15.

Containment isolation is addressed by a combination of the system's ITAAC and the containment isolation system ITAAC. The containment isolation valves are shown on the system figures. The verification of the design qualification of the motor-operated containment isolation valves will be verified by the basic configuration check in the system ITAACs as discussed under "General Provisions." In addition, in situ tests are required for containment

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isolation MOV and check valves in each system ITAACs. The containment isolation system ITAACs verifies that the containment isolation valves close on receipt of an isolation signal, verifies the electrical power assignment to the containment isolation valves, and verifies the failure response of the isolation valves, as applicable. Leak rate testing is addressed in the nuclear island structures.

The staff decided during the review of the ITAACs that the main control room (MCR) ITAACs would verify only the minimum inventory as derived from the emergency operations guidelines (EOGs), the requirements of RG 1.97, and PRA insights. Other controls, indications and alarms are identified in the system ITAACs and verified to exist in the MCR. The ability of these controls, indications, and alarms to function will be checked during operation of the system for the functional tests required by the system ITAACs. The operation of the system during the completion of the functional tests required in the system ITAACs will be conducted from the MCR. Therefore, it was decided that the verification that the system can be operated from the MCR need not be a separate ITAACs. The operation of equipment from the remote shutdown panel is addressed in the remote shutdown panel ITAACs.

14.3.2.3.2 Reactor Systems Task Group Review

The reactor systems task group had primary review responsibility for the reactor systems and core cooling systems in CDM Section 2.0. The group had secondary review responsibilities for those systems that could affect the operation of the reactor and core cooling systems.

The task group primarily used the SRP in its review of the CDM to determine the safety significance of SSCs. The group also used applicable rules and regulations, GDC, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from System 80+ safety and severe accident analyses, and took operating experience into account. The task group also used guidelines in the draft review guidance for the design control document as an aid for establishing consistency and completeness in its review of the systems. For selected portions of the review, the staff utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs."

The reactor systems task group review of the CDM systems was performed in a manner similar to as the plant

systems task group's review because the reactor and core cooling systems were primarily fluid systems. Thus, the group examined the systems for comprehensive and consistent treatment of the issues listed in Section 14.3.2.3.1 of this chapter, according to the safety significance of the respective systems being reviewed. The reactor systems task group found that many of the systems in this area of review were classified as safety related, and thus many of the characteristics and features of these systems had safety significance. This is reflected in a higher level of detail in the CDM for these systems.

The task group reviewed the CDM to verify that plant safety analyses, such as for core cooling, transients, overpressure protection, and anticipated transient without scram (ATWS), were adequately addressed. The task group reviewed the tables in CESSAR-DC Section 14.3 to determine that the important input parameters used in the transient and accident analyses were verified by the ITAACs. The task group worked with specialists in PRA and severe accident analyses to ensure insights and design features from these analyses were incorporated into the CDM. For the severe accident analyses in particular, the basis for the staff's review was the Commission guidance related to SECY-90-016 and SECY-93-087. For both PRA and severe accident analyses, although large uncertainties and unknowns may have been associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses were still selected for treatment in the CDM. The supporting information regarding the detailed design and analyses remained in the CESSAR-DC. For many of the design features, it was impractical to test their functionality. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown, was considered sufficient CDM treatment.

The staff determined that the detailed supporting information in the CESSAR-DC for the nuclear fuel system in the CDM, if considered for a change by a COL applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question. This supporting information comprises the fuel design, the control element assembly (CEA) design, and the initial core design. Thus, the staff has concluded that any changes to the initial reference design of these areas from that presented and evaluated in CESSAR-DC Chapter 4 will require prior NRC review and approval, except for the design features and design parameters in Tables 4.2-1 and 4.2-2 and changes covered by applicable NRC-approved topical reports. Furthermore, prior NRC review and approval will be required to change the NRC-approved analysis methods used to demonstrate conformance of the fuel design, the CEA design, and the initial core design to the design limits given in CESSAR-DC, except for those

changes to analysis methods that are covered by applicable NRC-approved topical reports. The specific fuel, CEA, and core designs presented in CESSAR-DC Chapter 4 will constitute, based on staff review and approval, an approved design that may be used for the COL first cycle core loading, without further NRC staff review. If any other core design is requested for the first cycle, the COL applicant or licensee will be required to submit for staff review those specific fuel, control rod, and core design analyses as described in CESSAR-DC Chapters 6 and 15.

Examples of the issues that the task group examined for treatment in Tier 1 included net positive suction head for key pumps (specified in the applicable systems), and intersystem LOCA (the design pressure of the piping of the systems that interface with the reactor coolant pressure boundary is specified in the design descriptions of the applicable systems). The task group also reviewed the ITAACs for consistency with the initial test program described in CESSAR-DC Chapter 14.

14.3.2.3.3 Electrical Task Group Review

The electrical task group had primary review responsibility for the electrical systems in CDM Section 2.6. The scope of the System 80+ electrical design includes the entire Class 1E portion of the electrical system as well as a major portion of the non-Class 1E electrical system and portions of the plant lighting system. The group also reviewed selected interface requirements. The group had some secondary review responsibilities for some systems using Class 1E power.

In establishing the top-level requirements for the electrical design, the staff used the GDCs of Appendix A to 10 CFR Part 50; and 10 CFR 50.49, "Environmental Qualification;" and 10 CFR 50.63, "Station Blackout," as its main bases. In addition, IEEE nuclear standards were used, as appropriate, to further establish top-level requirements. The staff also considered significant lessons learned from operating experience problems and insights gained from the PRA for the System 80+ design.

GDC 17, in part, requires that an onsite and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. It further requires that the onsite electric power system have independence and redundancy and that the electric power supplied by the offsite system be supplied by two physically independent circuits.

10 CFR 50.49 requires that certain electrical equipment be qualified for accident (referred to as "harsh") environments.

10 CFR 50.63 requires that a nuclear power plant be able to withstand and recover from a station blackout event.

IEEE 308, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," in conjunction with other related IEEE standards, establishes specific design criteria for nuclear power plant electrical systems and equipment.

The staff reviewed the System 80+ standard plant design to ensure, in part, that the certified design contains top-level design, fabrication, testing, and performance requirements for SSCs important to safety. Design descriptions and ITAACs were established to verify that these top-level (Tier 1) requirements (or design commitments) are met when the plant is built.

(1) Class 1E Electrical Systems

The System 80+ Class 1E electrical systems include: (1) portions of the Class 1E electrical power distribution system, (2) the emergency diesel generators, (3) the Class 1E dc power supplies, and (4) the Class 1E ac instrument and control power supplies. Using the regulations listed above, IEEE standards, operating experience, and PRA as its bases, ABB-CE established top-level design commitments for the Class 1E electrical systems of the System 80+ design to be included in the design descriptions and verified by ITAACs.

The top-level design commitments for the Class 1E electrical systems include design aspects related to

- equipment qualification for seismic and harsh environment
- redundancy and independence
- capacity and capability
- electrical protection features
- displays, controls, and alarms

Equipment Qualification

To ensure that the seismic design requirements of GDC 2 and the environmental qualification requirements 10 CFR 50.49 have been adequately addressed, "basic configuration" ITAACs were established for applicable systems to verify these design aspects of electrical equipment important to safety.

The design description identifies that Class 1E equipment is seismic Category I and that equipment located in a harsh environment is qualified. A "basic configuration" ITAACs was developed to include these areas.

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Redundancy and Independence

To ensure that the Class 1E electric systems meet the single-failure requirements of GDC 17 (and other GDC), ITAACs were established to verify the redundancy and independence of the Class 1E portion of the electrical design. For the electrical systems, ITAACs verified the Class 1E divisional assignments and independence of electric power by both inspections and tests. The independence is established by both electrical isolation and physical separation. The Class 1E divisional equipment is identified to aid in demonstrating the separation. (The detailed requirements are specified in the CESSAR-DC. For example, separation distances and identification are outlined in the CESSAR-DC.) These attributes are verified all the way to the electrically powered loads by a combination of the electrical system ITAACs and the ITAACs of the individual fluid, I&C, and HVAC systems which also cover the electrical independence and divisional power supply requirements.

Capacity and Capability

To ensure that the electrical systems have the capacity and capability to supply the safety-related electrical loads, ITAACs were established to verify the adequate sizing of the electrical system equipment and its ability to respond (e.g., automatically in the times needed to support the accident analyses) to postulated events. This includes the Class 1E portion and the non-Class 1E portion to the extent that it is involved in supporting the Class 1E system.

ITAACs are included to analyze the as-built electrical system and installed equipment (diesel generators, transformers, switchgear, batteries, etc.) to verify its ability to power the loads. In addition, the ITAACs also includes tests to demonstrate the operation of the equipment.

To ensure that the Class 1E portion of the electrical power system has the capability to respond to postulated events including LOCAs, loss of normal preferred power, and degraded voltage conditions, ITAACs were established to verify the initiation of the Class 1E equipment necessary to mitigate the event.

ITAACs are included to analyze the as-built electrical power system for its response to a LOCA, loss of voltage, combinations of LOCA and loss of voltage, and degraded voltage. In addition, tests are included to demonstrate the actuation of the electrical equipment in response to postulated events.

Electrical Protection Features

To ensure that the electrical power system is protected against potential electrical faults, ITAACs were established to verify the adequacy of the electrical circuit protection included in the design. Operating experience and NRC electrical distribution system functional inspections (EDSFIs) have indicated some problems with the short-circuit rating of some electrical equipment and breaker and protective device coordination.

ITAACs are included to analyze the as-built electrical system equipment for its ability to withstand and clear electrical faults. In addition, ITAACs are included to analyze the protection feature coordination to verify its ability to limit the loss of equipment due to postulated faults.

Displays, Alarms, and Controls

To help ensure that the electrical power system is available when required, ITAACs are included to verify the existence of monitoring and controls for the electrical equipment. The minimum set of displays, alarms, and controls is based on the emergency operations guidelines. In some cases, additional displays, alarms, and controls are specified based on special considerations in the design or operating experience.

ITAACs are included to inspect for the ability to retrieve the displays and alarms, and to control the electrical power system in the main control room and at locations provided for remote shutdown.

(2) Other Electrical Equipment Important to Safety

In addition to the Class 1E systems addressed above, other aspects of the electrical design were deemed to be important to safety and the top-level design commitments were included in the CDM.

Offsite Power

To ensure that the requirements of GDC 17 for the adequacy and independence of the preferred offsite power sources within the System 80+ scope were met, ITAACs were developed to verify the capacity and capability of the offsite sources to feed the Class 1E divisions, and the independence of those sources.

ITAACs are included to inspect the direct connection of the offsite sources to the Class 1E divisions and to inspect for the independence/separation of the offsite sources. Lightning protection and grounding features are inspected as part of the configuration ITAACs.

In addition, the design description includes "interface" requirements for the portions of the offsite power outside of the System 80+ scope, however no ITAACs are included for the interfaces. The interfaces define the requirements that the offsite portion of the design (that is out of scope) must meet to support and not degrade the in-scope design.

Containment Electrical Penetrations

To ensure the containment electrical penetrations (both those containing Class 1E circuits and those containing non-Class 1E circuits) do not fail due to electrical faults and potentially breach the containment, ITAACs were developed to verify that all electrical containment penetrations are protected against postulated currents greater than their continuous current rating.

ITAACs are included to inspect and analyze the electrical penetrations and their protection.

Combustion Turbine Generator

To ensure the availability of the combustion turbine generator (CTG) as an alternate ac source for station blackout events, ITAACs were developed to verify its inclusion in the design and its independence from other ac sources. In addition, the PRA has indicated that the CTG is relatively important from a risk perspective.

ITAACs are included to inspect and test the CTG and its auxiliaries.

Lighting

To ensure that portions of the plant lighting remain available during power failures, ITAACs were developed to verify the continuity of power sources for the lighting systems.

ITAACs are included to inspect and test the lighting and its power sources.

(3) Electrical Power For Non-safety Plant Systems

To ensure that electrical power is provided to support the non-safety-related plant systems, design descriptions cover portions of the non-Class 1E electrical systems. A basic configuration ITAACs verifies the functional arrangement and the Tier 1 design commitments for these areas.

14.3.2.3.4 Human Factors Task Group Review

The human factors task group had primary review responsibility for the main control room and remote shutdown

room described in CDM Section 2.12, "Human Factors." Additionally, the task group provided input to other task groups on the minimum inventory of displays, controls, and alarms for the control room and the remote shutdown room.

The basis for the task group's review in this area was a human factors engineering (HFE) program review model (PRM) developed by the staff (i.e., "HFE Program Review Model and Acceptance Criteria for Evolutionary Reactors"). The staff informed the Commission of the development of the PRM and associated acceptance criteria in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," dated August 27, 1992. In addition, the task group utilized the SRP in its review of the CDM. Other sources were applicable rules and regulations, RGs, USIs and GSIs, and operating experience.

The staff developed the HFE PRM, discussed in Section 18.1.3 of this report, to serve as a technical basis for the review of the design information proposed by ABB-CE for certification of the System 80+ control room and remote shutdown room design. The HFE PRM is (1) based upon currently accepted HFE practices, (2) well-defined, and (3) validated through experience with the development of complex, high-reliability systems in other industrial and military applications. The review model identifies the important HFE elements in a system development, design, and evaluation process that are necessary and sufficient requisites to successful integration of human factors in complex systems. The review model also identifies aspects of each HFE element that are key to a safety review, and describes acceptance criteria by which the HFE elements can be evaluated. The HFE PRM has eight program elements, each of which contain both general and more specific acceptance criteria. These elements are:

- Human Factors Engineering Program Management
- Operating Experience Review
- Functional Requirements Analysis
- Function Allocation
- Task Analysis
- Human-System Interface Design
- Plant and Emergency Operating Procedures
- Human Factors Verification and Validation

10 CFR Part 52 requires applicants for design certification to comply with the TMI requirements in 10 CFR 50.34(f)(2)(iii) for designing a control room that reflects state-of-the-art human factors principles. ABB-CE did not develop a final control room and remote shutdown room design before design certification because this is an area of

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rapidly changing technology. Instead, ABB-CE submitted a combination of detailed design information and design process requirements and acceptance criteria by which the main control room and remote shutdown room would be developed, designed, and evaluated. In lieu of having a completed control room design for review, the staff concluded that it could make its safety determination if ABB-CE submitted for certification an acceptable process for the design of the control room and remote shutdown room.

ABB-CE gave the staff detailed design information regarding several of the program review models elements (Human Factors Engineering Program Management, Operating Experience Review, Functional Requirements Analysis, Function Allocation, and Plant and Emergency Operating Procedure Development) which enabled the staff to complete a safety determination for those design elements. The staff's evaluation of these elements appears in Chapter 18 of this report. Additionally for those areas that were not completed (Task Analysis, Human-System Interface Design, and Human Factors Verification and Validation), ABB-CE submitted a description of the design processes and acceptance criteria necessary for the continuing detailed design effort in Chapter 18 of the CESSAR-DC and high-level requirements for the implementation of these processes in CDM Section 2.12.

To review the ITAAC proposed by ABB-CE, the staff was required to assess whether all appropriate elements of the HFE PRM were included. ABB-CE's ITAACs was compared to the major HFE PRM elements to determine if they were captured. As noted above, the following five elements were excluded from the ITAAC: Human Factors Engineering Program Management, Operating Experience Review, Functional Requirements Analysis, Function Allocation, and Plant and Emergency Operating Procedures. The first four exclusions were allowed because the staff had previously reviewed and found acceptable ABB-CE's detailed design documentation with respect to these HFE PRM elements. Procedural development was determined to be the responsibility of the COL applicant and was, therefore, designated as a COL action item. The remaining HFE PRM elements: Task Analysis, Human-System Interface Design, and Human Factors Verification and Validation are addressed in CDM Section 2.12.

In addition to the design process material, the staff requested that a minimum inventory of displays, controls, and alarms be developed through a task analysis of the operator actions necessary to carry out the EOGs and PRA critical actions. The staff's evaluation of the resulting minimum inventory encompassed a multi-disciplinary effort consisting of human factors, I&C, PRA, and plant, reactor, and electrical system engineering. The criteria used to deter-

mine acceptability of the inventory included assuring that (1) the scope of these items in the EOGs and PRA effort were adequately considered, (2) the task analysis was detailed and comprehensive, (3) RG 1.97, Category I variables for accident monitoring were included, and (4) important system indications and controls described in the Tier 1 system design descriptions necessary for transient mitigation were included.

The minimum inventory list for the control room was included in CDM Section 2.12.1, "Main Control Room." The controls and indicators required on systems to remotely shut down the reactor were included in CDM Section 2.12.2, "Remote Shutdown Room." Detailed supporting information is in Chapter 7 of the CESSAR-DC. The staff reviewed the individual systems that contained the sensors for the displays, controls, and alarms to ensure that standard ITAACs entries were used to verify their function. The design processes and acceptance criteria for I&C equipment CDM Section 2.5, particularly the verification and validation aspects of the I&C acceptance criteria, will verify proper operation of the I&C aspects of the equipment. Similarly, the design processes requirements for HFE in CDM Section 2.12, particularly the availability and suitability verification, and validation aspects of the control room and remote shutdown room, will verify proper design of the equipment for human factors considerations.

The staff conducted a complete and thorough review of the CDM material to ensure that the general criteria of the eight program elements in the HFE PRM were appropriately addressed in the Tier 1 CDM. The Tier 2 CESSAR-DC material contains more detailed guidelines and applicable guidance documents. The staff also reviewed the CESSAR-DC material to ensure that the specific acceptance criteria in the HFE PRM were appropriately addressed. The staff reviewed the CDM and CESSAR-DC Section 14.3.3 in accordance with the requirements in 10 CFR Part 52 and the guidance in SRMs related to design certification applications provided by the Commission, as discussed in the background part of Section 14.3 of this report.

The material in CESSAR-DC Chapter 18 provides design information and defines design processes for the four remaining HFE PRM elements that are acceptable for use in meeting the acceptance criteria in the CDM. However, the CESSAR-DC information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the System 80+ design for the control room is based on the design processes and acceptance criteria material as described in CDM 2.12 and Chapter 18 of the CESSAR-DC. Consequently, the staff indicated in

Chapter 18 and Section 1.11 of this report that any proposed changes to CESSAR-DC Sections 18.5, 18.6, 18.7, 18.8, and 18.9 constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval before implementation. These sections discuss

- "Functional Task Analysis," Section 18.5
- "Control Room Configuration," Section 18.6
- "Information Presentation and Panel Layout Evaluation," Section 18.7
- "Control and Monitoring Outside the Main Control Room," Section 18.8
- "Verification and Validation," Section 18.9

14.3.2.3.5 Radiation Protection Task Group Review

The radiation protection task group had primary review responsibility for the liquid, gaseous, and solid radioactive waste management systems, process and effluent radiological monitoring and sampling systems, and emergency response facilities in CDM Section 2.0; and the additional material in CDM Section 3.2, "Radiation Protection," applicable to multiple systems of the design; and selected site parameters. The staff's review of CDM Section 3.2 is in Section 14.3.3.2 of this chapter. The group had secondary review responsibility for all other ITAACs which addressed the plant radiation protection design or systems relied upon in the design-basis accident (DBA) dose assessment. These ITAACs included buildings, ventilation and filtration systems, primary containment, post-accident sampling system, and site parameters (atmospheric dispersion).

The task group primarily used the SRP in its review of the CDM to determine the safety significance of SSCs. Other sources included applicable rules and regulations, GDC, RGs, USIs and GSIs, NRC generic correspondence, and operating experience. The task group also used the draft review guidance for the design control document as an aid for consistency in its review of the systems.

The task group relied heavily on the material in CDM Section 3.2 during its review of the design. The design processes and acceptance criteria in this section were developed because ABB-CE did not submit sufficient information to stipulate the source terms needed to verify the design of the shielding, ventilation, and airborne radioactivity monitoring systems. Therefore, ABB-CE extracted the most important acceptance criteria for these design features from Chapter 12 of the SRP and put them into the DACs in CDM Section 3.2. A COL applicant or licensee must meet these criteria in the design of the plant, and the staff can audit the facility's design documentation to ensure that the criteria are met. The DACs are general

criteria which apply to the design of shielding, ventilation, and airborne radioactivity monitoring systems throughout the plant. Therefore, there are no references to the DACs in the ITAACs for the buildings and systems.

The group reviewed the ITAACs for the radioactive waste management systems to ensure that the ITAACs appropriately describe the systems features relied on by the staff in Chapters 11 and 12 of this report and to show that the plant can comply with the effluent limits of 10 CFR Part 20 and the effluent dose design objectives of Appendix I to 10 CFR Part 50. The group reviewed the ITAACs for the process and effluent radiological monitoring and sampling systems to ensure that effluents are monitored before being released to unrestricted areas, that information is submitted on radiation dose rates in the plant during normal operation and accidents, and that, the systems provide signals to safety equipment and alarms to warn plant personnel of changes in radiation dose rates. The group also reviewed the ITAACs for emergency response facilities to ensure that adequate facilities are provided for the technical support center (TSC) and operational support center (OSC), including space, data retrieval, and communications equipment, and a ventilation system to provide radiation protection.

The group had secondary review responsibility for the review of several ITAACs. The review of these ITAACs was focused on verifying design features and assumptions upon which the radiological dose consequence assessment of the design-basis accidents (DBAs) in this report is based. Important examples of these features and assumptions are discussed below. The maximum primary containment leakage rate will be verified in the nuclear island structures ITAACs. The minimum particulate removal efficiency of the particulate filters will be verified by the annulus ventilation system ITAACs, the fuel building ventilation system ITAACs, the containment purge ventilation system ITAACs, and the subsphere building ventilation system ITAACs. The capability of the annulus ventilation system to quickly draw a slight negative pressure on the annulus will be verified by the ITAAC for that system. The minimum particulate and radioiodine removal efficiencies of the filters in the control room and TSC ventilation systems will be verified by the control complex ventilation system ITAACs. The size of flow-limiting orifices in the letdown line will be verified by the chemical and volume control system ITAACs. The location and separation of the two control room emergency air intakes will be verified by the nuclear island structures ITAACs. The capability of the control room intake monitors to select and close the most contaminated emergency air intake will be verified by the ITAACs for that system. The capability of the gaseous waste management system to withstand an internal hydrogen explosion will be

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verified by the ITAACs for that system. The capability of the gaseous waste management system to withstand collapse in a safe-shutdown earthquake will also be verified by the ITAACs for that system. In addition, the meteorological dispersion values assumed in the accident analyses were identified as parameters for a site in CDM Table 5.0, "Site Parameters."

14.3.2.3.6 Structural Task Group Review

The structural task group had primary review responsibility for building structures, chemical engineering systems, site parameters, piping DACs, reactor vessel (RV), and the legends for figures. The piping DAC contained in CDM Section 3.1 is discussed in Section 14.3.3.1 of this chapter. The task group had secondary review responsibilities for other systems as they related to MOVs, check valves, hydrostatic tests, and seismic and safety classification of systems, and for other structural aspects of systems. The task group was composed of reviewers who had experience in structural, mechanical, materials, and chemical engineering. In addition, the task group was augmented by a specialist in MOVs, check valves, and pumps; a specialist in seismic and safety classification; and a specialist in chemical engineering.

The task group primarily consulted rules and regulations to review the top-level commitments in the CDM. Other sources included RGs, SRP guidelines, and PRA insights from System 80+ safety and severe accident analyses and operating experience. For selected portions of the review, the staff also adhered to policy discussions by the Commission in the SRM related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance in the SRM related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." In addition, the task group reviewed the Tier 1 submittals (including the design description, figures, and ITAACs) of the design using the guidelines in the draft review guidance for the CDM as an aid for establishing consistency and completeness.

The task group reviewed the design description for those assigned systems to ensure that the certified design was consistent with the NRC regulations and policy decisions as discussed in SECY-93-087. The task group reviewed the Tier 1 material to assess whether a conclusion could be reached that the ITAACs were necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Atomic Energy Act, and the Commission's rules and regulations.

The task group reviewed the design descriptions, figures, ITAACs, and the CESSAR-DC for consistency. The task group reviewed the CDM for all the systems to ensure consistency with the seismic and safety classification described in Section 3.2 of the CESSAR-DC. The task group ensured that the seismic classification of the system as described in the design description and the ASME Code Class boundaries of the system as depicted on the figures was consistent with CESSAR-DC Section 3.2.

The task group reviewed the ITAACs tables for consistency with the preoperational tests specified in Chapter 14 of the CESSAR-DC. The task group reviewed the tests identified in the ITAACs tables to determine whether those tests have been appropriately included in CESSAR-DC Chapter 14 and also whether the preoperational tests have been adequately incorporated into ITAACs. The task group also reviewed tests in Chapter 14 of the CESSAR-DC or in the ITAACs that would require an analysis to convert preoperational test conditions to accident conditions to ensure that the methodology for performing the analysis was specified adequately. In addition, the task group reviewed all systems ITAACs to ensure that selected issues were adequately and consistently treated in the CDM through the use of the standard ITAACs entries for basic configuration, hydrostatic test, MOVs, and check valves.

The task group used the following general approach in reviewing the design descriptions, figures, and ITAACs and for establishing what information should reside in each tier. The certified design (design description) should contain top-level design, fabrication, testing, and performance requirements for SSCs important to safety. ITAACs are established, in part, to verify that these top-level (Tier 1) design, fabrication, testing, and performance requirements are adhered to when the plant is built.

Although the establishment of what specific information was to be included in the design description was essentially a matter of judgment, the draft review guidance provides some guidance for consistency in certain areas regarding what information should be in which tier as well as whether an inspection, test, or analysis is required to be performed. The draft review guidance also provided a basis for the staff's judgment in selecting which tier the information should reside in and why an ITAACs was deemed necessary. These areas include component welding, equipment seismic qualification, pumps, valves, and piping systems. The basis for selecting these areas was its importance to safety as well as its past experience with construction and operating problems.

Design descriptions and ITAACs were developed and grouped by systems and building structures. These Tier 1

requirements for systems and building structures are typically verified by inspections, tests, and analyses specified in the system ITAACs. For example, system-specific performance tests are typically conducted to demonstrate that the system can perform its intended function. For building structures, the structural capability is typically verified by performing an analysis to reconcile the as-built data with the structural design bases for each safety-related building.

(1) Review of Components

For components, the verification of design, fabrication, testing, and performance requirements are partially addressed in conjunction with the specific system ITAACs. For example, a test is typically performed to verify the ability of a motor-operated valve to close under design-basis fluid conditions. However, performance tests are not practical for verifying certain component design requirements such as its seismic design or safety classification. Therefore, ITAACs have been developed to verify certain areas in which performance tests are impractical. These areas include seismic design qualification and fabrication (i.e., welding) of components. The ITAACs for seismic design qualification and fabrication of components are established on a generic basis in the general provisions for verifying the basic configurations of systems rather than on an individual component basis.

The staff reviewed the Tier 1 treatment of the design qualification and fabrication of components to ensure that the issues were verified by ITAACs as discussed below:

Fabrication of Components

A basic configuration check (system) is required in each individual system ITAACs. The configuration check includes an inspection of the welding quality for all ASME Code Class 1, 2, and 3 piping systems. A hydrotest is also required in each system ITAACs for ASME Code Class 1, 2, and 3 piping systems to verify that, in the process of fabricating the overall piping system, the welding and bolting requirements for ensuring the pressure integrity have been met. The methods to be used by the COL applicant or licensee to verify the acceptability of the welds are discussed in the CESSAR-DC in the sections applicable to the specific component or structure.

Design Qualification of Components

- (a) **Safety Classification** — The safety classification of SSCs are described in each system's design description. The functional drawings identify the boundaries of the ASME Code classification that are applicable to the safety class. The piping DAC

includes a verification of the design report to ensure that the appropriate code design requirements for the system's safety class have been implemented.

- (b) **Mechanical and Electrical Equipment (including I&C)** — A basic configuration check (system) is required in each individual system ITAACs. The configuration check includes an inspection of the as-built equipment (including anchorages) and a review of the qualification records to verify that the equipment in its as-built condition is seismically qualified. The material in CESSAR-DC Section 3.10.1 provides detailed supporting information for the CDM regarding the methods to be used by the COL applicant or licensee for the dynamic qualification of equipment. This material, if considered for a change by an applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval before implementation of the change. This material supporting the CDM is discussed further in Section 3.10 of this report.
- (c) **Valves** — The design qualification of valves is verified in conjunction with the basic configuration check for mechanical equipment as discussed above. Specifically, for MOVs, a special inspection is required as a part of the basic configuration check to verify the records of vendor tests that demonstrate the ability of MOVs to function under design conditions. In addition, in-situ tests are required for MOVs and check valves in each system ITAACs. These tests will be performed during the initial test program. The material in CESSAR-DC Section 3.9.6.2.2 provides detailed supporting information for the CDM regarding the methods to be used by the COL applicant or licensee for the design, qualification, and testing of MOVs to demonstrate their design-basis capability. This material, if considered for a change by an applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. This material supporting the CDM is discussed further in Section 3.9.6 of this report.
- (d) **Piping** — The overall piping design, including the effects of high-energy line breaks and the application of leak-before-break (as applicable), is verified in conjunction with the piping DACs. The as-built piping system is required to be reconciled with the design commitments. The material in CESSAR-DC Section 3.12 provides detailed supporting information.

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tion for the CDM regarding the analysis methods and design criteria to be used by the COL applicant or licensee to complete the piping design. This material, if considered for a change by an applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval before implementation of the change. This material supporting the CDM is discussed further in Section 3.12 of this report.

(2) Review of Structural Design Integrity

The scope of structural design covers the major structural systems in the System 80+ plant, including the RV, ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, nuclear island structures, turbine building, component cooling water (CCW) heat exchanger structures, diesel fuel storage structures (DFSSs), and radwaste building). The RV, piping systems, and primary containment are included because they provide the defense-in-depth principle for nuclear plants. The major building structures house those systems and components that are important to safety.

In establishing the top-level requirements for structural design, the staff used the GDC of 10 CFR Part 50 (Appendix A) as its basis. The primary general design criteria pertaining to the major structural system design are GDC 1, "Quality Standards and Records"; GDC 2, "Design Bases for the Protection Against Natural Phenomena"; GDC 4, "Environmental and Dynamic Effects Design Basis"; GDC 14, "Reactor Coolant Pressure Boundary"; GDC 16, "Containment Design"; and GDC 50, "Containment Design Basis."

- GDC 1 requires, in part, the need for structures, systems, and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- GDC 2 requires, in part, the need to design structures, systems, and components important to safety to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods without loss of capability to perform their safety functions, including the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.
- GDC 4 requires, in part, the need to protect structures, systems, and components important to safety from dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that

may result from equipment failures and from events and conditions outside the nuclear power unit.

- GDC 14 requires, in part, the need for the reactor coolant pressure boundary to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
- GDC 16 requires, in part, the need for the reactor containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment.
- GDC 50 requires, in part, the need for the reactor containment structure, including access openings and penetrations to be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

Using these GDC as its basis, the staff established the following top-level attributes to be verified by ITAACs:

- pressure boundary integrity (GDC 14, 16 and 50)
- normal loads (GDC 2)
- seismic loads (GDC 2)
- flood, wind, and tornado (GDC 2)
- rain and snow (GDC 2)
- pipe rupture (GDC 4)
- codes and standards (GDC 1)

In addition, to ensure that the final as-built plant conforms to the certified design, ABB-CE submitted ITAACs to reconcile the as-built plant with the structural design basis. A summary of the top-level structural design requirements for the major structural systems that are verified by the structures and systems in CDM Section 2.0 and the piping design information in CDM Section 3.1 is given below.

Pressure Boundary Integrity

To ensure that the applicable requirements of GDC 14, 16, and 50 have been adequately addressed, ITAACs were established to verify the pressure boundary integrity of the RV, piping, and primary containment for the System 80+. GDC 16 and 50 apply to the primary containment, and GDC 14 applies to the RV and the reactor coolant pressure boundary piping systems. The pressure integrity for these major structural systems are needed to ensure the defense-in-depth principle.

For the RPV and piping, hydrostatic tests performed in conjunction with the ASME Boiler and Pressure Vessel Code, Section III are required by ITAACs. For the primary containment, a structural integrity test is required by ITAACs to be performed on the pressure boundary components of the primary containment in accordance with the ASME Boiler and Pressure Vessel Code, Section III. Because the requirements of GDC 14, 16, and 50 do not apply to the other nuclear island structures, turbine building, CCW heat exchanger structures, DFSSs, and radwaste building ITAACs were not required for verifying the pressure integrity of these other buildings.

Normal Loads

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAACs were established to verify that the normal and accident loads have been appropriately combined with the effects of natural phenomena.

For piping systems, ITAACs require an analysis to reconcile the as-built piping design with the design-basis loads (which include the appropriate combination of normal and accident loads). For the RV, the fabrication is performed primarily in the vendor's shop, where adherence to design drawings is tightly controlled. Therefore, ITAACs for the as-built reconciliation of normal loads with accident loads for the RV were deemed inappropriate. Instead, ITAACs verify that the ASME Code-required reports exist to document that the RV has been designed, fabricated, inspected, and tested to code requirements to ensure adequate safety margin.

Similarly, for safety-related buildings, ITAACs require an analysis for reconciling the as-built plant with the structural design-basis loads (which include the combination of normal and accident loads with the effects of natural phenomena). The analysis results are to be documented in a structural analysis report, the scope and contents of which are described in the CESSAR-DC. The staff determined that the design of some structures did not require verification by ITAACs, based on its safety significance. In particular, these ITAACs apply only to safety-related structures and are not applicable to the radwaste and turbine buildings.

Seismic Loads

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAACs were established to verify that the safety-related systems and structures have been designed to seismic loadings. Component qualification for seismic loads is addressed by ITAACs that were

established for verifying the basic configuration of systems.

As discussed above for normal loads on piping systems and the RV, ITAACs require an analysis to reconcile the as-built piping design with the design-basis loads (which include seismic loads). For the RV, ITAACs for the as-built reconciliation of seismic loads for the RV were deemed to be inappropriate, as previously discussed. Instead, ITAACs verify that the ASME Code-required reports exist for the RV, ensuring that the RV has been designed, fabricated, inspected, and tested to ASME Code requirements.

For safety-related buildings, ITAACs require an analysis for reconciling the as-built plant with the structural design-basis loads (which include seismic loads). The analysis results are to be documented in a structural analysis report, as discussed above. These ITAACs apply only to safety-related structures and are not applicable to the radwaste and turbine buildings. However, because the leakage path for fission products includes components within the turbine building, the turbine building is required to withstand the effects of a safe-shutdown earthquake. Therefore, ITAACs were established to verify that, under seismic loads, the collapse of the radwaste and the turbine buildings will not impair the safety-related functions of any structures or equipment located adjacent to or within the radwaste and turbine buildings.

For non-seismic Category I SSCs, the need for ITAACs to verify that their failure will not impair the ability of nearby safety-related SSCs to perform their safety-related functions was assessed. Because the design detail and as-built and as-procured information for many non-safety-related systems (e.g., field-run piping and balance-of-plant systems) are not required for design certification and the spatial relationship between such systems and seismic Category I SSCs cannot be established until after the as-built design information is available, the non-seismic to seismic (II/I) interaction cannot be evaluated until the plant has been constructed. Accordingly, the design criteria for assuring acceptable II/I interactions and a commitment for the COL applicant to (1) describe the process for completion of the design of balance-of-plant and non-safety-related systems to minimize II/I interactions, and (2) provide procedures for an inspection of the as-built plant for II/I interactions have been specified as a COL action item in the CESSAR-DC.

Flood, Wind, Tornado, Rain, and Snow

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAACs were established to

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verify that the safety-related systems and structures have been designed to withstand the effects of natural phenomena other than those associated with seismic loadings. The effects include those associated with flood, wind, tornado, rain, and snow.

These loadings do not apply to the RV, the ASME Code Class 1, 2, and 3 piping systems and components, nor the primary containment, because they are all housed within the safety-related buildings. For safety-related buildings, ITAACs require an analysis for reconciling the as-built plant with the structural design basis loads (which include the flood, wind, tornado, rain, and snow loads). Based on their safety significance, these ITAACs apply only to safety-related structures and are not applicable to the radwaste and turbine buildings.

For flooding, site parameters are specified that require the maximum flood level and ground water level be below the finished plant grade level. ITAACs also require inspections to verify that divisional flood barriers and water-tight doors exist, and penetrations (except for water-tight doors) in the divisional walls are sealed up to the internal and external flood levels. In addition, for safety-related buildings, flood barriers are established up to the finished plant grade level to protect against water seepage, and flood doors and flood barrier penetrations are provided with flood protection features.

Pipe Break

To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAACs were established to verify that the safety-related SSCs have been designed to the dynamic effects of pipe breaks. Component qualification for the dynamic effects of pipe breaks is addressed by ITAACs established for verifying the basic configuration of systems.

For the RV, ITAACs that verify the basic configuration of the RPV system require an inspection of the critical locations that establish the bounding loads in the LOCA analyses for the RPV to ensure that the as-built areas do not exceed the postulated break areas assumed in the LOCA analyses.

In addition, ITAACs have been established to verify by inspections of as-built, high-energy pipe break mitigation features and of the pipe break analysis report that safety-related SSCs be protected against the dynamic and environmental effects associated with postulated high-energy pipe breaks. ITAACs to verify pipe break loads are not required for the turbine and radwaste build-

ings either because they are not safety-related structures or there are no high-energy lines located within the structure.

Codes and Standards

To ensure that the applicable requirements of GDC 1 have been adequately addressed, ITAACs were established to verify that appropriate codes and standards were used in the design and construction of safety-related systems and components. In general, the staff considered those codes and standards endorsed by the regulations under 10 CFR 50.55a in determining which codes and standards were appropriate for Tier 1 verification. The ASME Boiler and Pressure Vessel Code, Section III, for Code Class 1, 2, and 3 systems and components was established as the code for the design and construction of System 80+ piping systems and the RV. For safety-related building designs, the staff based its safety findings on audits of System 80+ design calculations which relied on specific codes and standards. These codes and standards are in CESSAR-DC Section 3.8, and were identified in Section 3.8 of this report as material that, if considered for a change by an applicant or licensee that references the certified System 80+ design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change.

Inspections will be conducted as a part of ITAACs to verify that ASME Code-required documents exist which demonstrate that the RV, piping systems, and containment pressure boundaries have been designed and constructed to their appropriate code requirements. For other ASME Code components and equipment, code compliance will be verified in conjunction with the quality assurance programs and by the authorized inspection agency as required by the ASME Boiler and Pressure Vessel Code.

As-Built Reconciliation

To ensure that the final as-built plant structures are built in accordance with the certified design as required by 10 CFR Part 52, structural analyses will be performed which reconcile the as-built configuration of the plant structures with the structural design bases of the certified design. The structural analyses will be documented in structural analysis reports. Structural analysis reports will be verified in conjunction with ITAACs for the primary containment and the other nuclear island structures, radwaste building, CCW heat exchanger structures, DFSSs, and turbine building. The detailed supporting information on what is required for an acceptable analysis report is in CESSAR-DC Chapter 3.

Similarly, an as-built analysis will be performed for piping systems using the as-designed and as-built information. ITAACs will verify the existence of acceptable final as-built piping stress reports which conclude that the as-built piping systems are adequately designed.

For component qualification, tests or analyses, or a combination of tests and analyses will be performed for seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) to demonstrate that the as-built equipment and associated anchorages are qualified to withstand design-basis dynamic loads without loss of safety function. These tests and analyses will be performed as a part of ITAACs to verify the basic configuration of the system in which the equipment is located.

14.3.2.3.7 Instrumentation and Controls (I&C) Task Group Review

The I&C task group's primary review responsibilities included a review of the CDM for I&C systems involving core protection and control in CDM Section 2.5, "Instrumentation and Control," other miscellaneous I&C systems, the I&C material in CDM Sections 2.5 and 2.7.25 applicable to multiple systems of the design, and selected interface requirements. The group's secondary review responsibilities involves ESF systems, reactivity control systems, and other systems using I&C equipment.

The figures in CDM Section 2.5 depict both safety-related and non safety-related systems of the design. The block concept was used for developing the system control interface diagram that was needed for depicting the configuration of the I&C system architecture. The I&C design in the CESSAR-DC was described to the level of control functional blocks and, therefore, the configuration in the CDM was to the same level.

The task group reviewed CDM entries to confirm that the safety-related I&C system conformed to the protection systems requirements of 10 CFR 50.55a(h), as well as the quality standards and records requirements of GDC 1; the protection against natural phenomenon requirements of GDC 2; the environmental and dynamic effects requirements of GDC 4; the instrumentation and control requirements of GDC 13; the control room requirements of GDC 19; the protection system design requirements of GDC 20; the protection system reliability and testability requirements of GDC 21; the protection system independence requirements of GDC 22; the protection system failure modes requirements of GDC 23; the protection system requirements for reactivity control malfunctions of GDC 25; and the protection against anticipated operational occurrences

requirements of GDC 29. To comply with the criteria of 10 CFR 50.55a(h) ("Criteria for Protection Systems for Nuclear Generating Stations") and IEEE Standard 279-1971, the CDM entries were reviewed considering the following design issues:

- General functional requirements for the system
- Single failure criterion
- Quality of components and modules (hardware and software)
- Equipment qualification
- Channel integrity and channel independence
- Classification of equipment
- Isolation devices
- Single random failure
- System inputs
- Capability for sensor checks, tests and calibration
- Channel bypasses, operating bypasses, indication of bypasses, and access to means for bypassing
- Completion of protective action once initiated
- Manual initiation
- Information presentation
- Identification

Standard ITAACs entries for several attributes of the I&C system were developed and used for basic configuration, divisionalized power supply, electrical isolation and physical separation (independence), and control room and remote shutdown room configuration. For those systems reviewed that were not safety-related systems, appropriate criteria from the SRP applicable to those systems were used.

The staff used the SRP guidelines to review the I&C design information in the System 80+ CDM and SAR to confirm that both the safety-related and non-safety-related I&C systems complied with the appropriate acceptance criteria of the SRP. The staff also used the Commission guidance contained in an SRM of July 15, 1993, related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," dated April 2, 1993. The staff informed the Commission of the development of the DAC in this area in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," dated August 27, 1992.

ABB-CE did not submit complete design information in the CESSAR-DC for the microprocessor and digital control technology aspects of the I&C system design of the System 80+ because the technology in this area is rapidly evolving and it is, therefore, important that the certified design description and ITAACs not "lock in" a design which

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could be obsolete at the time of construction. The process to complete the I&C design, with appropriate acceptance criteria, is specified in CDM Section 2.5, with detailed supporting information in CESSAR-DC Chapter 7. Therefore, these process-related I&C design acceptance criteria are discussed in this part of the chapter, and are not addressed separately. The issues discussed in that material include the plant protection system (PPS), engineered safety features (ESFs) system, supporting systems, hardware and software development, electromagnetic compatibility, instrument setpoint methodology, environmental qualification of I&C equipment, and I&C system diversity and defense-in-depth considerations. Since the process in these CDM sections to complete the design applies to both safety-related and non-safety-related I&C systems, the staff relied heavily on the information in those CDM sections in its reviews of the I&C systems.

CDM Section 2.5 has multiple entries addressing three key issues associated with the I&C design: the design of the safety systems, the development and qualification processes for I&C systems, and design features that provide diverse backup as protection against common-mode failures in the safety systems. These issues and their relationships to other systems of the design are illustrated in the block diagrams in CDM Section 2.5, which show the safety system logic and control, a depiction of the integrated hardware and software development process for I&C systems, and diagram the interfaces of the safety systems with other I&C systems in the design. Detailed supporting information for the CDM is in CESSAR-DC Chapter 7.

CDM Section 2.5.1, "Plant Protection System," and 2.5.2, "Engineered Safety Features-Component Control System," contain material for the safety system logic and control. The plant protection system (PPS) and the ESFs component control system (ESF-CCS) integrate the automatic decision-making with trip logic functions and manual initiation functions associated with the safety actions of the safety-related systems. Safety-related trip logic and monitoring of the PPS reside in the PPS and ESF-CCS equipment. The PPS and ESF-CCS equipment comprises microprocessor-based, software-controlled signal processors that perform signal conditioning, setpoint comparison, trip logic, system initiation and reset, self-test, calibration, and bypass functions.

The acceptance criteria in the appropriate subsections of the CDM describe (1) design processes and acceptance criteria to be used for safety-related systems using programmable microprocessor-based control equipment, (2) a program to assess and mitigate the effects of electromagnetic interference on I&C equipment, (3) a program to establish setpoints for safety-related instrument channels, (4) a program to qualify safety-related I&C equipment for

in-service environmental conditions, and (5) the hardware and software development processes to be used in the design, testing, and installation of I&C equipment. The discussion which follows addresses the considerations made in reviewing the entries for this section of the CDM.

The primary function of this development process is to implement the functional I&C requirements described in the CDM and the CESSAR-DC for the systems in the System 80+ design. The decomposition of the functional system (PPS, ESF-CCS, etc.) requirements to specific computer hardware and software components to perform the various tasks is accomplished by means of the structured design process described below.

In the CDM, ABB-CE describes the design process for hardware and software development, design commitments, the inspections, tests, and analysis to be performed to verify that the design is consistent with the commitments, and the appropriate acceptance criteria against which the design will be judged. These ITAACs describe attributes of the process to be used to develop the software, as well as attributes of the final software product. The ITAACs entries for software and hardware describe the following design stages within the design process:

- Requirements analysis
- Design
- Implementation or coding
- Test
- Installation and Checkout
- Operation and Maintenance
- Retirement

These stages are based on generic activities that represent software and system life-cycle processes, and encompass expert opinions, existing national standards, and input from ongoing efforts in the revisions of national and international standards for computer systems in nuclear power plants.

The CDM and CESSAR-DC contain criteria which describe the method to develop detailed plans and procedures that will guide the design process throughout the above life-cycle design stages listed above. The ITAACs provide the acceptance criteria for verifying the design through these design stages; the CESSAR-DC adds the set of guidelines and standards that will provide more detailed criteria for the development of the design. The CDM has been written to incorporate the most important and general aspects (top-level requirements) from the standards. The set of standards and criteria in the CESSAR-DC encompass the guidance for generating the plans that will be used in the computer software and hardware design process for the computer design throughout the life cycle. These plans are described below.

ABB-CE provided a software development plan (SDP), NPX80-SQP-0101, "Nuplex 80+ Software Program Manual" describing the complete software development process, including hardware integration. The purpose of the SDP is to establish the organization and authority structure for the design, the procedures to be used, and the interrelationships between and among major activities.

ABB-CE's SDP consists of a software quality assurance (SQA) plan, a software configuration management (SCM) plan, a software verification and validation (V&V) plan, and an software operations and maintenance (O&M) plan. The elements of a software management plan are incorporated into the SQA plan, the V&V plan, and the SCM plan. The integration plan, installation plan, and training plan are to be developed during the remainder of the planning stage. Although the software safety plan was submitted independent of the SDP, the elements of the SSP are incorporated into the SDP.

The SQA plan describes the software-specific activities that are to be performed and controlled in addition to the approved QA plan (in accordance with 10 CFR Part 50 (Appendix B), "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants") for the total System 80+ design. The SQA plan establishes the criteria under which the other software development plans will be generated.

The SCM plan provides the means to identify software products, control and implement changes, and record and report change implementation status.

The V&V plan provides a means by which software can be evaluated through its life cycle to ensure that the software system requirements are accurately translated into the final software product. Each software module will be verified by an organization that is independent of the organization that developed the software module.

The O&M plan contains the procedures required to ensure that the software will be operated correctly and that the quality of the software will be maintained. ABB-CE's O&M plan specifies the organizational requirements and error reporting requirements that the COL licensee shall observe during the operation phase and the retirement phase of the software life cycle. The O&M plan has error-reporting requirements and forms to ensure that digital system errors are reported in a manner that is consistent with the configuration management guidelines in the SCM plan section of the SDP.

The NRC will inspect the ITAAC activities completed by the COL applicant to verify conformance with the require-

ments at several stages during the digital control system design process or stage of the life cycle. The documents which demonstrate satisfactory implementation of the ITAACs will be available for inspection during the NRC audit at the completion of each of the several stages. The stages or phases described by ABB-CE are shown in Figure 7.1 in this report. The NRC audit and the COL applicant conformance review points are also shown in Figure 7.1. These stages correspond with the phases described by ABB-CE in the CDM. The actual stages, including the conformance review and audit points, will be determined for each of the software products to be developed when design implementation is scheduled to begin.

At each stage, the COL applicant must verify that the design development complies with the certified design process and that the detailed design developed through that stage is in conformance with the certified design. Upon completing ITAAC activities for each stage, the COL applicant will certify to the NRC that the stage has been completed, and that the design and construction completed up through that stage complies with the certified design. Although not required, the COL applicant should satisfactorily complete the ITAAC activities at each stage before proceeding to the next stage of the design development process. Failure to successfully complete the ITAAC at a stage, as determined by the conformance review or the NRC audit, may require repeating an earlier stage in the ITAACs or changing the system design, or both. The NRC staff will keep track of any open issues which require resolution for each stage in the ITAACs. Significant open issues which are not resolved could lead the NRC staff to conclude that the ITAACs had not been satisfactorily completed.

The ITAACs address electromagnetic compatibility (EMC), which is the process to ensure that I&C equipment is able to function properly when subjected to an electromagnetic environment. An EMC compliance plan to confirm the level of immunity to electrical noise is included in the design, installation, and testing of I&C equipment. The plan is structured on the basis that EMC of I&C equipment is verified by factory testing and site testing of both individual components and interconnected systems to meet EMC requirements.

The setpoint methodology portion of the ITAACs describes the process to ensure that setpoints for initiation of safety-related functions are determined, documented, installed, and maintained. The process (the instrument setpoint methodology) establishes a program for specifying requirements for documenting the bases for selection of trip setpoints, accounting for instrument inaccuracies, response testing, and replacement of instrumentation.

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The equipment qualification portion of the ITAACs describes the process to ensure that qualification of safety-related I&C equipment is able to complete its safety-related function under the environmental conditions that exist up to and including the time the equipment has finished performing that function. An equipment qualification program is established that ensures qualification specifications consider conditions that exist during normal, abnormal, and design-basis accident events in terms of their cumulative effect on equipment performance for the period up to the end of equipment life.

The diversity and defense-in-depth portion of the ITAACs address the concern that software design faults or other initiating events common to redundant, multi-divisional logic channels of I&C protection systems could disable significant portions of the plant's safety functions at the moment when these functions are needed to mitigate an accident; diverse backup features are provided for the primary automatic logic. Diversity is provided in the form of hard-wired backup for reactor trip, diverse display of important process parameters, defense-in-depth arrangement of equipment, and other equipment diversity.

The staff reviewed the System 80+ CDM material thoroughly to ensure that the SRP guidelines and Commission guidance for I&C design were appropriately addressed in both the CDM and the CESSAR-DC. The staff evaluated the analysis methods, design procedures, acceptance criteria, and related ITAACs that are to be used for the completion and verification of the System 80+ I&C design. The CESSAR-DC material has more detailed guidelines and applicable documents. The staff reviewed the CDM and CESSAR-DC Section 14.3.2 in accordance with the requirements in 10 CFR Part 52 and the guidance in Commission SRMs related to design certification applications, as discussed in the background part of Section 14.3 of this report.

Selected material in CESSAR-DC Chapter 7 provides design information and defines design processes that are acceptable for use in meeting the I&C acceptable criteria in the CDM. This material includes design information regarding hardware and software changes, commercial dedication, equipment qualification, and electromagnetic compatibility. However, the CESSAR-DC information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the System 80+ design for I&C systems is based on the design processes and acceptance criteria material in the CDM and the CESSAR-DC. Consequently, the staff indicated in Chapter 7 of this report that any proposed changes to the appropriate CESSAR-DC sections constitutes an unreviewed safety

question and, therefore, must be submitted to the NRC for review and approval before implementation.

The certified design description and design development process continue for the lifetime of the plant. Any safety-related software that is changed or added after plant startup is required to either be developed using the certified design development process described in the CDM, or the licensee must submit a design process (together with the design bases) description that will produce a digital system of the same or higher quality than the original certified design process, consistent with the CDM. The licensee will be required to use an approved software change procedure based upon the certified design development process for the operation stage of the life cycle.

14.3.2.4 Approval of the CDM for Structures and Systems

The staff performed a multidisciplinary review, utilizing several task groups, of the SSCs of the System 80+, in accordance with 10 CFR Part 52 and the guidance in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple CDM and CESSAR-DC submittals to the staff, as listed under "Background" in Section 14.3 of this chapter.

On the basis of the task group reviews, the staff concludes that the top-level design features and performance characteristics of the SSCs important to safety in the System 80+ design are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design features and performance characteristics can be adequately verified by the ITAACs submitted by ABB-CE. Therefore, the staff concludes that the ITAACs in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the SSCs important to safety in a facility that references the design have been constructed and will operate in accordance with the design certification and applicable regulations.

14.3.2.5 Initial Test Program (ITP)

CDM Section 2.11 consists of a high-level commitment to an ITP and a description of the program and major program documents (i.e., a site-specific startup administrative manual, test specifications, and test procedures). CESSAR-DC Section 14.2 contains a complete description

of the ITP, and the staff's evaluation of the ITP is in Section 14.2 of this chapter.

The staff reviewed this CDM section for consistency with the guidelines in the SRP and RG 1.68, "Initial Test Program for Water-Cooled Nuclear Power Plants." RG 1.68 describes the general scope and depth of testing that is acceptable to the staff for conduct of preoperational and startup testing as part of the ITP.

The key facets of the ITP are described in the Tier 1 CDM to ensure that subsequent changes in the conduct of the ITP cannot be initiated unilaterally by the COL applicant. This ITP is described in Tier 1 because of the essential role of a test program in the verification that SSCs have been constructed and will perform satisfactorily in service. The Tier 1 description requires that the ITP be performed under suitably controlled conditions and processes. The development of test procedures, conduct of the tests, and safe execution of the test program, are important considerations in ensuring that the as-built facility is in accordance with the design certification and applicable regulations. Thus, the staff will have the confidence that the ITP will be implemented effectively, so that the appropriate testing methodologies and associated programmatic controls for testing plant systems will be ensured.

A corresponding ITAACs for this design description is not required for several reasons:

- (1) The Tier 1 certified design material consists of a high level commitment to an ITP, and a description of the program and major program documents that constitute an acceptable ITP (i.e., a site-specific startup administrative manual, test specifications, and test procedures). The specific testing necessary to verify design features and performance aspects of the design is delineated in the system-specific ITAACs.
- (2) The ITP covers a broader spectrum of time than the ITAACs. While ITP pre-operational testing shall be completed before fuel load, the ITP startup and power ascension testing will be conducted after fuel load. As the ITP involves testing post-fuel load, it is not appropriate to define associated ITAACs entries as Part 52 specifies that the ITAACs will be completed before fuel load.

In summary, the top-level ITP commitments in the CDM ensure that suitable controls are imposed over the preoperational and startup testing programs, which provide reasonable assurance that the facility can be operated without undue risk to the public. The staff concludes that the ITP information in the CDM is acceptable.

14.3.2.6 DSER Issues

The "Certified Design Material for the ABB-CE System 80+," was under staff review at the time the DSER was issued. The staff stated that the final evaluation would appear in the this report. This was designated as DSER Open Item 14.1-1.

Many sections of the DSER contained preliminary comments on the CDM for the ABB-CE System 80+ design. These preliminary comments were documented in the sections of the DSER based on their relationship to the detailed design information in various CESSAR-DC sections. The preliminary comments in the DSER were intended to discuss the philosophy of development of the Tier 1 CDM, establish early staff positions on the material, and indicate the status of development of the material.

ABB-CE resolved all of the staff's comments in various revisions to the Tier 1 CDM, and provided revised detailed supporting design information in various CESSAR-DC amendments as discussed in the background part of Section 14.3 of this chapter. On the basis of the revised material in the CDM and CESSAR-DC, the CDM development process, criteria, and methodology described in CESSAR-DC Section 14.3, and the review process discussed here in this chapter, the staff concludes that these issues were adequately addressed where appropriate in the CDM. Therefore, DSER Open Item 14.1-1 is resolved.

Table 14.1 lists open issues in the DSER that specifically identified the Tier 1 material, and that were resolved as discussed above. This list of issues considered for treatment in the CDM is not all inclusive. Since issuing the DSER, the staff considered many issues for treatment in the CDM, using the general approach and criteria discussed herein. Other specific issues considered for treatment in the CDM may also be discussed in the appropriate sections of this report.

The issues in Table 14.2 were classified in the DSER as ITAAC Confirmatory Items. They were also resolved based on the revised CDM, supporting CESSAR-DC information, the discussion in CESSAR-DC Section 14.3, and the discussion in this section of this chapter.

14.3.3 Additional Certified Design Material

This section of the System 80+ CDM provides additional certified design material for design and construction activities that are applicable to more than one system. There are two entries in this CDM section. The entries describe design-related processes and associated design acceptance criteria (DACs) for the System 80+. The design description for each entry describes its scope and

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Table 14.1 ITAAC open items in the DSER

Item Number	Description of Item
3.7-1	The applicant is required to complete the seismic analyses of all Category I structures and update the CESSAR-DC to include Tier 1 and 2 information.
5.4.3.1-2	The applicant is required to include a boron mixing test in the ITAACs program to demonstrate a satisfactory boron mixing under natural circulation.
6.5-2	The calculated containment spray system pump minimum NPSH could not be compared to required NPSH since no specific pump has been selected. (ITAAC)
6.7.1-2	The applicant is required to specify the required flow capacity of RCGVS and submit an ITAAC.
6.7.2-2	The applicant is required to specify the required flow capacity of RDS, and submit an ITAAC.
7.1.2-1	The applicant is required to submit ITAAC/DAC for final I&C.
7.7.1.21-1	The conformance of the I&C software development process verification and validation program to ANSI/IEEE-ANS-7-4.3.2-1982 is an unresolved ITAAC item.
8.3.1.13-1	The applicant is required to submit an ITAAC for containment electrical penetrations.
8.3.1.16-1	The applicant is required to submit an ITAAC to verify breaker coordination design commitments.
8.3.1.17-1	The applicant is required to submit an ITAAC to verify equipment fault current interrupting capability design commitments.
9.5.2-1	The staff will review the ITAAC to ensure PABX, intraplant PA system, and intraplant sound powered telephone system are independent.
14.1-1	ITAAC will be reviewed in this report.
19.1.2.4-1	The applicant is required to submit a list of items that could be included under DAC, ITAACs, or RAP requirements.
20.1-3	The staff will review PRA and ITAACs submittals when they are received and the applicant is required to provide the requirements for conducting a walkdown on "as built" plants.
20.4-2	Regarding GL 88-15, the applicant has committed to a number of studies and analyses of the electric power-systems. The staff will confirm the adequacy of these actions as part of the ITAACs.

Table 14.2 ITAAC confirmatory items

Item Number	Description of Item
4.2.7-1	The applicant is required to provide the fuel design acceptance criteria as ITAACs to be certified for System 80+ fuel design.
4.4.6.3-1	The staff will confirm that an ITAAC is developed to include testing and calibration of the inadequate core cooling (ICC) system.
6.3.8-1	The applicant is required to commit to a preoperational testing program and an ITAACs program for safety injection.

applicability to the System 80+ design. Amplifying information on CDM Section 3.0 is in CESSAR-DC Section 14.3.3. The material in this section of the CDM applies to the individual systems of the System 80+ design contained in CDM Section 2.0, and the staff's review of the material in CDM Section 2.0 is in Section 14.3.2 of this chapter. The staff's safety evaluation for each design area in which the DACs are used is contained in the section of this report applicable to the area.

Design Acceptance Criteria (DAC)

ABB-CE did not submit design and engineering information for some areas of the System 80+ design at a level of detail customarily reviewed by the staff in making a final safety determination. ABB-CE submitted less detailed information in these areas because ABB-CE believed they were either areas of rapidly changing technology, and it would have been detrimental to freeze the details of the design many years before an actual plant was ready to be constructed, or because they were areas for which ABB-CE did not have sufficient as-built or as-procured information to complete the final design. Areas of rapidly changing technology included control room and RSS design (human factors) and advanced instrumentation and controls. Areas dependent on as-built or as-procured information included piping design and radiation shielding, ventilation, and airborne monitoring design. The staff provided its views on the DAC to the Commission on February 19, 1992, in SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews."

The design information and appropriate design methodologies, codes, and standards in the CESSAR-DC, together

with the design descriptions and DACs, are sufficiently detailed to serve as an adequate basis for the staff to make a final safety determination regarding the design, subject only to satisfactory design implementation and verification of the DACs by the COL applicant or licensee. The DACs are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination in support of the System 80+ design certification. The acceptance criteria for the DACs are inspectable, testable, or subject to analysis using pre-approved methods, and must be verified as a part of the ITAACs performed to demonstrate that the as-built facility conforms to the certified design. Thus, the acceptance criteria for DACs are specified together with the related ITAACs in the Tier 1 material, and both are part of the design certification. The DACs and the ITAACs, when conformed to, ensure that the completed design and as-constructed plant conforms to the design certification. The material in the CESSAR-DC for each of the DAC areas includes, as appropriate, sample calculations or other supporting information to illustrate methods that are acceptable to the staff for meeting Tier 1 DAC commitments.

The structure of each area in which DACs are used is the same as for the other areas of the design that are verified by ITAAC. The structure consists of three parts: the Tier 1 design description, the corresponding DACs, and the Tier 2 supporting information in the CESSAR-DC for the DACs. The staff has based its safety findings for the areas in which DACs are used on the Tier 2 information specified in the CESSAR-DC, including applicable design methodologies, codes, and standards, contingent on verification that the design has been properly implemented according to the Tier 1 design descriptions and the corresponding DACs.

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14.3.3.1 Piping Design DACs

The piping design aspects of the System 80+ design are given in CESSAR-DC Appendix 3.9A and (together with the associated DACs in CDM Section 3.1, "Piping Design"), are evaluated in Section 3.12 of this report. ABB-CE did not submit the complete design information in this design area before design certification because the piping design was dependent upon as-built and as-procured information. Instead, ABB-CE submitted the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. ABB-CE submitted amplifying information in CESSAR-DC Section 14.3.3.1 regarding CDM Section 3.1 in this area. The material in CDM Section 3.1 applies to System 80+ piping systems classified as nuclear safety-related as specified in the Tier 1 material for the individual systems in CDM Section 2.0 and to certain high-energy non-nuclear safety systems. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

The staff used the SRP guidelines and RGs to evaluate the piping design information in the System 80+ CDM and CESSAR-DC and performed a detailed audit of the piping design criteria, including sample calculations. The staff evaluated the adequacy of the structural integrity and functional capability of safety-related piping systems. The review was not limited to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, 2, and 3 piping and supports, but covered buried piping, instrumentation lines, the interaction of non-seismic Category I piping with seismic Category I piping, and any safety-related piping designed to industry standards other than the ASME Code. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAACs that are to be used for the completion and verification of the System 80+ piping design. The staff evaluated both CDM and CESSAR-DC information regarding the applicable codes and standards, analysis methods to be used for completing the piping design, modeling techniques, pipe stress analyses criteria, pipe support design criteria, high-energy line break criteria, and leak-before-break (LBB) approach applicable to the System 80+.

The material in CDM Section 3.1 provides the design process to develop the piping for the nuclear safety-related (seismic Category I) systems of the System 80+ design. Piping systems that must remain functional during and following an SSE are designated as seismic Category I and are further classified as ASME Code Class 1, 2, or 3. The piping systems and their components are designed and constructed in accordance with the ASME Code requirements identified in the individual systems of the System

80+ design. The CDM ensures that the piping systems will be designed to perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. The material in the CDM section also addresses such consequential effects of pipe ruptures as jet impingement, potential missile generation, and pressure and temperature effects.

ABB-CE specified four ITAACs in the CDM to ensure the design process for piping systems was as stated in the design description. The first ITAAC requires that an as-built piping stress report exists that documents the results of an as-built reconciliation analysis confirming that the final piping system has been built consistent with the assumptions used in the ASME Code certified stress report. As used in this report, an ASME Code certified stress report is the design document required by ASME Code, Section III, Subarticle NCA-355-0. The report verifies that the as-constructed piping system is consistent with the certified design commitments. This verification assures that modification of any document used for construction from the corresponding document used for design analysis has been reconciled with the certified stress report discussed above. This documentation will become part of the certified stress report. The detailed supporting information on what is required for an acceptable analysis report is in CESSAR-DC Chapter 3.

The second ITAAC specified in the CDM requires that an ASME Code certified stress report exists to ensure that the ASME Code Class 1, 2, or 3 piping systems are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design-basis loads. The specific contents and requirements of the certified stress report are in the ASME Code. A certified piping stress report gives assurance that requirements of the ASME Code, Section III for design, fabrication, installation, examination, and testing have been complied with, and that the design complies with the design specifications.

The third ITAAC requires that an LBB evaluation report exists which documents that LBB acceptance criteria are complied with for the as-built piping and piping materials. This verification is needed because in the design certification stage, bounding limits were established using piping evaluation diagrams (PEDs) and preliminary analysis results to establish a window of acceptable piping stress values for selected piping materials. The COL applicant is required to verify that these values are complied with using actual material properties and final piping configurations. In CESSAR-DC Appendix 3.9A, the applicant describes the contents of the LBB evaluation report and

tells how the COL applicant is to reconcile the as-built piping data with the LBB design assumptions.

The fourth ITAACs requires that a pipe break analysis report exists documenting that SSCs required to be functional during and following an SSE have adequate high-energy pipe break mitigation features. As discussed in CESSAR-DC Chapter 3, the pipe break analysis report specifies the criteria used to postulate pipe breaks and the analytical methods used to perform pipe breaks, and confirms the adequacy of the results of the pipe break analyses. This verification gives assurance that the high-energy line break analyses have been completed and comply with the following certified design commitments. For postulated pipe breaks, the Pipe Break Analysis Report shall confirm that: (1) piping stresses in the containment penetration area shall be within their allowable stress limits, (2) pipe whip restraints and jet shield designs shall be capable of mitigating pipe break loads, (3) loads on safety-related SSCs shall be within their design load limits, and (4) SSCs are protected or are qualified to withstand the environmental effects of postulated failures. The Pipe Break Analysis Report shall conclude that, for each postulated piping failure, the reactor can be shut down safely and maintained in a safe, cold shutdown condition without offsite power.

The staff conducted a complete and thorough review of the System 80+ CDM to ensure that the SRP guidelines for piping design were appropriately addressed in both the Tier 1 CDM and the CESSAR-DC. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAACs that are to be used for the completion and verification of the System 80+ piping design. The Tier 2 CESSAR-DC material contains more detailed guidelines and applicable documents. The staff reviewed the CDM and CESSAR-DC Section 14.3.3.2 in accordance with the requirements in 10 CFR Part 52 and the guidance in SRMs related to design certification applications provided by the Commission. The staff reviewed information in submittals as listed in the "Background" of Section 14.3 of this chapter.

Selected material in CESSAR Chapter 3 and Appendix 3.9A provides design information and defines design processes that are acceptable for use in meeting the piping DACs in the CDM. However, the CESSAR-DC information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the System 80+ design for piping systems is based on the design processes and acceptance criteria material in the DACs and the CESSAR-DC. Consequently, the staff stated in Section 3.12 of this report that any proposed changes to selected aspects of these piping design processes described

in the appropriate CESSAR-DC sections constitute an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval before implementation.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the piping design aspects of SSCs important to safety in the System 80+ are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features, and performance characteristics can be adequately verified by the DACs, provided by ABB-CE. Therefore, the staff concludes that the DACs in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the piping design aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.3.2 Radiation Protection DACs

The radiation protection aspects of the System 80+ design are discussed in CESSAR-DC Chapter 12, "Radiation Protection," and together with the associated DACs in CDM Section 3.2, "Radiation Protection," are evaluated in Chapter 12 of this report. ABB-CE did not submit the complete design information in this design area before design certification because the radiation shielding design and the calculated concentrations of airborne radioactive material were dependent upon as-built and as-procured information of plant systems and components. Therefore, ABB-CE was not able to describe the System 80+ radiation source terms (i.e., the quantity and concentration of radioactive materials contained in or leaking from plant systems) in sufficient detail to allow the staff to verify the adequacy of the shielding design, ventilation system design, or the design and placement of the airborne radioactivity monitors. Instead, ABB-CE submitted the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. ABB-CE submitted amplifying information regarding the processes in this area in CESSAR-DC Section 14.3.3.2. This material in CDM Section 3.2 applies to the radiological shielding, ventilation design, and airborne radioactivity monitoring systems of the nuclear island structures, turbine building, component cooling water heat exchanger structures, and radwaste building.

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The implementation of the process and the design is the responsibility of the COL applicant or licensee.

The acceptance criteria in the DACs are taken from the acceptance criteria in the applicable section of Chapter 12 of the SRP. The analysis methods and source-term assumptions specified in the DACs are consistent with approved methods and assumptions listed in the SRP. The SRP is the basis for the staff's safety review of the System 80+ design. Therefore, demonstrating that the final design complies with these DACs with the methods and assumptions specified in the CDM ensures that the as-built System 80+ design complies with the applicable acceptance criteria of the SRP and the associated regulations and staff technical positions. The staff informed the Commission of the development of the DACs in this area for the lead evolutionary plant designs in SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR)," dated May 28, 1992.

The DACs in the CDM information address the verification of the plant radiation shielding design and the plant airborne concentrations of radioactive materials (e.g., the ventilation system and airborne monitoring system designs). The DACs require the COL applicant to calculate radiation levels and airborne radioactivity levels within the plant rooms and areas to verify the adequacy of these design features during plant construction (concurrently with the verification of the ITAACs). The plant rooms and areas to which the DACs apply are given in the figures in CDM Section 3.2. Detailed supporting information is in CESSAR-DC Chapter 12.

The criteria in CDM Table 3.2-1, Items 1 and 2, ensure that the radiation shielding design (either that provided for by the plant structures, or design permanent or temporary shielding) is adequate to ensure that the maximum radiation levels in plant areas are commensurate with the area's access requirements so radiation exposures to plant personnel can be maintained as low as reasonably achievable (ALARA) during normal plant operations and maintenance. Item 3 in CDM Table 3.2-1 ensures that adequate shielding is provided for those areas of the plant that may require occupancy to permit an operator to aid in the mitigation of or the recovery from an accident. Item 4 in CDM Table 3.2-1 ensures that the plant provides adequate containment and ventilation flow rates to control the concentrations of airborne radioactivity to levels commensurate with the access requirements of areas in the plant. Item 5 in CDM Table 3.2-1 ensures that once the concentrations of airborne radioactivity are determined per Item 4 above, the required airborne monitors are provided in the appropriate locations in the plant. Item 6 in CDM Table 3.2-1 ensures that the contribution to the radiation dose

to a member of the public (off site) from direct and scattered radiation from on-site sources of radiation will be a small fraction of the EPA dose limit in 40 CFR Part 190.

The staff conducted a complete and thorough review of the ABB-CE System 80+ CDM to ensure that the SRP guidelines for radiation protection design were appropriately addressed in both the CDM and the CESSAR-DC. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAACs that are to be used for completing and verifying the System 80+ radiation protection design. The Tier 2 CESSAR-DC material contains more detailed guidelines and applicable documents. The staff reviewed the CDM and CESSAR-DC Section 14.3.3.2 in accordance with the requirements in 10 CFR Part 52 and the guidance in Commission SRMs related to design certification applications.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the radiation protection aspects of SSCs important to safety in the System 80+ are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features, and performance characteristics can be adequately verified by the DACs provided by ABB-CE. Therefore, the staff concludes that the DACs in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the radiation protection aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.4 Interface Requirements

The requirements for interfaces for a design are contained in 10 CFR 52.47(a)(1)(vii-ix). An applicant for design certification is required to submit (1) the interface requirements to be met by those portions of the plant for which the application does not seek certification, (2) justification that compliance with the interface requirements is verifiable through inspection, testing, or analysis, and the method to be used for verifying interface requirements, and (3) a representative conceptual design for those portions of the plant for which the application does not

seek certification. The staff evaluated these interface requirements and the System 80+ design in the appropriate sections of this report.

ABB-CE defined the interface between the systems of the design and the site-specific systems to be at the walls of the nuclear island structure, turbine building, radwaste building, diesel fuel storage building, and component cooling water heat exchanger building, as depicted in Figure 1.2-1 of CESSAR-DC. This section of the CDM specifies the safety-significant interface requirements for those portions of the certified design that interface with site-specific portions of the design, and specifies the safety-significant systems that are completely or partially outside the scope of the System 80+ certified design. CESSAR-DC Section 1.9 contains detailed information on the interface requirements for the design. The interface requirements define the most important design attributes and performance characteristics that must be met by the site-specific, out-of-scope portion of the plant so that the site-specific portion of the design is in conformance with the certified design. The site-specific portions of the design are those portions of the design, such as the design of the ultimate heat sink, that are dependent on characteristics of the site.

The appropriate inspections, tests, and analyses to demonstrate compliance with the interface requirements for the site-specific portion of the design are evaluated in the review of an application for a combined license under Subpart C of 10 CFR Part 52.

ABB-CE submitted information discussing the interface requirements in the CDM and in CESSAR-DC Section 14.3.4. ABB-CE provided acceptable interface requirements in CDM Section 4.0, and in the appropriate systems in CDM Section 2.0. This information was based on the information in the various sections of the CESSAR-DC, and is evaluated by the staff in the appropriate sections of this report. In CDM Section 4.0, ABB-CE stated that the development of ITAACs for the interface requirements will be similar in nature to the development of ITAACs in CDM Section 2.0. The staff concludes that this is an acceptable justification — that compliance with the interfaces is verifiable through ITAACs — and the process described in CESSAR-DC Section 14.3 is an acceptable methodology for verifying the interface requirements. ABB-CE submitted acceptable representative conceptual designs in the CESSAR-DC that enabled the staff to complete its review of the design, as discussed in the appropriate sections of this report.

Therefore, on the basis of this discussion, the staff concludes that the interface information submitted by ABB-CE

conforms to the requirements in 10 CFR 52.47(a)(1)(vii-ix), and is acceptable.

14.3.5 Site Parameters

The requirements for site parameters for a design are contained in 10 CFR 52.47(a)(1)(iii). An applicant for design certification is required to submit the site parameters used in the design, and an analysis and evaluation of the design in terms of these parameters. The site parameters are specified in both the CDM Section 5.0 and Chapter 2 of the CESSAR-DC, and the analysis and evaluation of the design is contained in the applicable sections of the CESSAR-DC. The staff evaluated these parameters and the design in the appropriate sections of this report.

Site parameters are specified in this section of the CDM for establishing the bounding parameters to be used in the selection of a suitable site for a facility referencing the System 80+ certified design. Because they were used in bounding evaluations of the certified design, they define the requirements for the design that must be met by a site to ensure that a facility built on the site remains in conformance with the design certification. The demonstration that the site parameters are met at a given site is accomplished in conjunction with an application and issuance of a combined license under Subpart C of 10 CFR Part 52.

ABB-CE submitted information discussing the site parameters in the CDM and in CESSAR-DC Section 14.3.5. ABB-CE provided acceptable site parameters postulated for the certified design in CDM Section 5.0 and in the appropriate sections of the CESSAR-DC. The appropriate sections of the CESSAR-DC also contained an acceptable analysis and evaluation of the design in terms of these parameters, and the staff found the design acceptable in the related sections of this report. Therefore, the staff concludes that the site parameter information submitted by ABB-CE conforms to the requirements of 10 CFR 52.47(a)(1)(iii), and is acceptable.

14.3.6 Summary

The staff reviewed the ABB-CE System 80+ CDM and CESSAR-DC Section 14.3 in accordance with the requirements in 10 CFR Part 52 and the guidance in Commission SRMs related to design certification applications. The staff reviewed information contained in submittals to the staff, as listed under "Background" in Section 14.3 of this chapter.

On the basis of its review of the material in the CDM and its review of the selection methodology and criteria for the

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development of the CDM contained in CESSAR-DC Section 14.3, the staff concludes that the top-level design features and performance characteristics of the System 80+ SSCs important to safety are appropriately described in the CDM, and are acceptable.

Further, these top-level commitments can be adequately verified by the ITAACs and additional certified design material provided by ABB-CE. Therefore, in the appropriate parts of Section 14.3 of this chapter, the staff concludes that the CDM are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed, and the acceptance criteria met, a facility referencing the certified design will be constructed and will operate in conformity with the design certification and applicable regulations.

14.3.7 Confirmatory Issues

In the advance FSER issued by NRC letter dated March 8, 1994, the staff stated that it would verify that the final CDM appropriately incorporated treatment of the following issues: ACRS comments on piping and radiation protection DAC and ACRS comments on fires, floods, and tunnels for the lead evolutionary plant design; treatment of the OSC; delineation of scope and interface requirements; ITP; and a functional description of the ac independent water addition system for containment spray (including supporting information in CESSAR and preoperational testing). This was part of Confirmatory Item 14.3.7-1. ABB-CE incorporated the treatment of various issues in a subsequent revision to the CDM, together with supporting information in CESSAR-DC Amendment W. The staff finds this acceptable.

In the advance FSER, the staff also stated that it would verify that a discussion of the acceptance criteria for control room design and the I&C design, and a discussion of the Initial Test Program had been incorporated into a future amendment of CESSAR-DC Section 14.3. This item was actually part of Confirmatory Item 14.3.7-1 discussed above, but was incorrectly identified as Confirmatory Item 14.3.7-2. ABB-CE incorporated this discussion in CESSAR-DC Amendment W. The staff finds this acceptable.

Therefore, based on the above, Confirmatory Item 14.3.7-1 is resolved.

In the advance FSER, the staff stated that the CDM will be the portion of the design certified by the System 80+ design certification rulemaking. This information will be difficult to change after certification, and therefore the CDM information must reflect the design accurately. Consequently, the staff formed an independent review

group to ensure consistency of the CDM and the CESSAR-DC in January 1994. The staff's approval of the certified design was contingent on the completion of that review and satisfactory resolution of their comments. This item was actually Confirmatory Item 14.3.7-2, but was incorrectly identified as Confirmatory Item 14.3.7-3. The independent review group completed its review in March 1994, and its comments were provided to ABB-CE after review by the staff. ABB-CE provided satisfactory resolutions to all the comments in a subsequent revision to the CDM and provided adequate supporting information in amendments to CESSAR-DC. The staff finds this acceptable. This resolves Confirmatory Item 14.3.7-2.

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