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**LICENSING ASSESSMENT OF THE CANDU PRESSURIZED
HEAVY WATER REACTOR**

Preliminary Safety Information Document

June 1977

Work Performed Under Contract No. EY-76-C-02-2477

United Engineers and Constructors, Incorporated
Philadelphia, Pennsylvania

TECHNICAL INFORMATION CENTER
UNITED STATES DEPARTMENT OF ENERGY

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LICENSING ASSESSMENT
OF
THE CANDU PRESSURIZED
HEAVY WATER REACTOR

PRELIMINARY SAFETY INFORMATION DOCUMENT

VOLUME II

Prepared for the

ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

(Under Contract No. EY-76-C-02-2477)

June 1977

By

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PRELIMINARY SAFETY INFORMATION DOCUMENT

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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 Introduction and Summary

1.1.1 Purpose

ERDA has requested United Engineers & Constructors (UE&C) to evaluate the design of the Canadian natural uranium fueled, heavy water moderated (CANDU) nuclear reactor power plant to assess its conformance with the licensing criteria and guidelines of the U.S. Nuclear Regulatory Commission (USNRC) for light water reactors. A separate study is underway to evaluate the applicability of LWR General Design Criteria to the HWR design characteristics.

This assessment was used to identify cost significant items of nonconformance and to provide a basis for developing a detailed cost estimate for a 1140 MWe, 3-loop Pressurized Heavy Water Reactor (PHWR) located at the Middletown, USA Site.

ERDA specifically instructed UE&C to follow a conservative approach even where the applicability of U.S. licensing criteria was not clear. This was done so as to obtain in this study an upper limit regarding the cost involved in adapting the PHWR to the U.S. licensing environment. For this reason, this study does not reflect, in all areas, the compensating cost factors associated with areas where Canadian licensing requirements are more conservative or where different design solutions were followed to meet the same licensing intent.

It should further be noted that the information available to UE&C concerning the PHWR was limited to preliminary design descriptions that were of limited scope and depth.

This report then identifies areas of nonconformance and, in most instances, provides suggested design alterations which would permit the PHWR to meet current USNRC requirements. Identification of nonconformance to USNRC licensing criteria reflects different approaches to design philosophy but is not intended to imply that the design reviewed is less safe than a plant licensed in the United States.

1.1.2 Organization

The technical description of the HWR is organized according to the guidelines of Regulatory Guide 1.70, Revision 2, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition". Some deviation from this format occurs where the format requires site-specific data or where insufficient information exists on the HWR to provide a detailed discussion.

Accordingly, our understanding of how the plant works and lack of information in some areas necessitates that this document be preliminary in nature, and therefore will be referred to henceforth as the "Preliminary Safety Information Document", (PSID).

The interpretation of acceptability of a HWR design by the USNRC is centered around the General Design Criteria (GDC) in Title 10, Chapter I, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, and the Division 1 Regulatory Guides used to implement the General Design Criteria. Volume 1 of this document presents a detailed discussion of specific nonconformance to the General Design Criteria of 10 CFR 50.

1.1.3 Definitions

All system descriptions in this document are organized according to the following format, with the headings as defined:

- a) Function - describes what function the system is intended to perform.
- b) Description - describes the equipment and how it carries out its intended function.
- c) Design Bases - provides the operating conditions and design margins under which the equipment is designed to operate. Where design bases are not known, the subsection will be left blank. All design bases will be of Canadian origin unless otherwise indicated.
- d) Safety Evaluation
 - A. For safety related systems, the format is organized as follows:
 - (1) Applicable NRC Criteria - identifies the applicable NRC licensing criteria against which the system is to be assessed.
 - (2) Assessment - evaluates the conformance of the system to the applicable NRC Criteria. In some instances, comments are expressed which are not specific licensing requirements, but which reflect current U.S. nuclear industry practice. All such comments will be identified as not explicitly required for licensing.
 - (3) Recommended design change - outlines the minimum design change which appear to be necessary to a system or structure to meet U.S. licensing criteria. System design change descriptions are provided in outline form to allow

estimating the cost of the redesign, and no effort is made to provide a detailed redesign.

B. For non-safety related systems, no evaluation is provided.

However, in some instances, comments are provided which are not intended as requirements for change but rather to assist the reader to understand current U.S. practice, with respect to that kind of equipment.

C. For sections which discuss analyses, the safety evaluation identifies U.S. licensing requirements only, and no attempt is made to verify or evaluate the Canadian safety analyses for any particular postulated event.

1.2 Summary

The PHWR design contains certain features that will require significant modifications to comply with USNRC siting and safety requirements. The most significant of these features are the reactor vessel design; control systems; quality assurance program requirements; seismic design of structures, systems, and components; and providing an inservice inspection program capability. None of these areas appear insolvable with current state-of-the-art engineering or with upgrading of the quality assurance program for components constructed outside of the USA.

In order to be licensed in the U.S., the entire reactor assembly would have to be redesigned to comply with ASME Boiler & Pressure Vessel Code, Section III, Division 1 and Division 2, as discussed in Sections 4.3 and 5.1 of this report.

A summary matrix at the end of Volume 1 identifies compliance of the systems and structures of the PHWR plant with the USNRC General Design Criteria. The matrix further identifies the estimated incremental cost to a 600 MWe PHWR that would be required to license the plant in the U.S. Further, the matrix identifies whether or not the incremental licensing cost is size dependent and the relative percentage of the base direct cost of a Canadian sited plant.

1.3 General Plant Description

1.3.1 Principal Characteristics

The Heavy Water Reactor (HWR) represents a unique commercially available nuclear power reactor system which has come into world-wide prominence. The objective is to produce a power reactor having a high neutron economy with the material and manufacturing resources commercially available.

The design concept has dictated the major features of the system. The use of natural uranium which was available domestically in Canada and on the world markets required the use of heavy water to permit the economic use of the natural uranium fuel. A pressure tube design was adopted when it became evident that pressure vessels would become very large and complex as the sizes of reactors increased. Fuel bundles of short length and on-power refueling were adopted. These permit an economic burn-up of the fuel to be realized despite the relatively small excess reactivity in a natural uranium-heavy water system under equilibrium conditions.

The HWR possesses two design features which contribute to its efficient and economical performance. Both features have been proven to work well in practice.

The first feature is the fact that a pressure tube or calandria tube may be replaced relatively easily.

The other feature is the capability, while the reactor continues to operate, to detect and remove fuel which has suffered a defect in the Zircaloy cladding and which has become a source of contamination in the coolant. This capability to remove a source of contamination without delay helps greatly in keeping radioactivity from the primary coolant system at a low level and aids maintenance.

The features which make this possible are the system of individual fuel channels, each with their own inlet and outlet feeder pipes; a monitoring system which measures the activity in the feeder pipes coming from each channel; the fueling machines which can attach to each fuel channel and remove the fuel without interrupting the reactor power; and the relatively short length, 50 cm, of the fuel bundles which facilitates on-power refueling.

A general arrangement of the principal components and systems and their operation which forms the HWR nuclear power system is shown in Figures 1.3-1 and 1.3-2. Figure 1.3-3 is a cutaway view of the Reactor Building showing principal components.

1.3.2 Facility Description

1.3.2.1 Plot Plan

1.3.2.1.1 Function

The relative orientation of buildings on a site is determined by optimizing engineering efficiency of the plant with safety considerations.

1.3.2.1.2 Description

A typical single unit HWR plant layout is shown in Figure 1.3-4. As can be seen, the typical plant utilizes once-through cooling and the containment-turbine building orientation is such as to minimize length of main steam piping. The typical plant features on-site burial of solid radioactive waste. The fuel oil storage tanks are arranged close to the structures to minimize piping from the tank to the equipment.

1.3.2.1.3 Design Bases

Unknown.

1.3.2.1.4 Safety Evaluation

1.3.2.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the relative orientation of the buildings on the site:

- GDC 2 - Design bases for protection against natural phenomena.
- GDC 3 - Fire protection.
- GDC 4 - Environmental and missile design bases.
- GDC 5 - Sharing of structures, systems and components.

In addition, the Federal Water Pollution Control Act has a significant impact.

1.3.2.1.4.2 Assessment

In order to be licensed in the U.S., re-orientation of the buildings as well as additional structures are necessary. In particular, the orientation of the turbine building would have to be such as to minimize the probability of turbine missile impact on the containment or safety related equipment in the Service Building. In addition, the emergency diesel generators would have to be housed in a separate Seismic Category I structure.

Although burial of low-level radioactive waste is not expressly prohibited by the Code of Federal Regulations, it is standard practice in the U.S. for such waste to be shipped off-site. The location of the fuel oil storage tanks to the main plant structures should be sufficient to minimize damage to the remainder of the plant as a result of fire or explosion.

The Federal Water Pollution Control Act requires that the facility design be such as to demonstrate that the national goal of eliminating pollutants in

streams, navigable waters and oceans is maintained. The owner of the facility must prove that operation of the plant does not have an adverse impact on bodies of water. Such environmental regulations are generally interpreted in the U.S. as a requirement for cooling towers and eliminating once-through cooling.

1.3.2.1.4.3 Recommended Design Change

- GDC 2 - Provide separate diesel generator building to meet seismic design requirements.
- GDC 3 - Move fuel oil storage tanks away from plant structures as well as provide embankments around the tanks to contain spills.
- GDC 4 - Re-orient the turbine building such that the containment building is centered along the turbine axis.
- GDC 5 - Applies only to multi-unit plant sites. No design change is necessary since this assessment is for a single unit plant.

Federal Water- Provide redundant cooling towers.
Pollution
Control Act

When the above changes have been effected, the plant arrangement would be as shown in Figure 1.3.4a.

1.3.2.2 Reactor Building

1.3.2.2.1 Function

The Reactor Building is a Seismic Category I, prestressed post-tensioned concrete structure, which serves as missile protection for the reactor and associated radioactive systems as well as a final barrier for an uncontrolled release of radioactivity following an accident.

1.3.2.2.2 Description

The Reactor Building is a prestressed post-tensioned concrete structure with an epoxy liner which houses the Calandria, Primary Heat Transport System components and associated cooling and purification systems. Reactor Building general arrangement drawings are shown in Figure 1.3-5 through 1.3-15.

The Reactor Building is divided into normally accessible areas and normally inaccessible areas to facilitate maintenance and refueling during full power operation.

Systems internal to the Reactor Building are discussed in appropriate sections of the PSID. All internal walls in the Reactor Building are designed to withstand differential pressures encountered during loss of coolant or steam line break accidents.

1.3.2.2.3 Design Bases

Refer to Table 6.2-1.

1.3.2.2.4 Safety Evaluation

1.3.2.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Reactor Building.

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 3 - Fire protection.
- GDC 4 - Environmental and missile design bases.
- GDC 16 - Containment design.
- GDC 38 - Containment heat removal.
- GDC 39 - Inspection of containment heat removal system.
- GDC 40 - Testing of containment heat removal system.
- GDC 41 - Containment atmosphere cleanup system.
- GDC 42 - Inspection of containment atmosphere cleanup system.
- GDC 43 - Testing of containment atmosphere cleanup system.
- GDC 50 - Containment design bases.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 52 - Capability for containment leakage rate testing.
- GDC 53 - Provisions for containment inspection and testing.
- GDC 60 - Control of release of radioactive materials to the environment.

1.3.2.2.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with in the design and construction of the containment. The program would have to be expanded to include all phases of construction including the material manufacturer and suppliers.

The containment would also have to be designed for tornado wind loadings and negative pressure as required by GDC 2. (Refer to Volume 1, Section 3.3 discussion on GDC 2). Internal walls and floors separating accessible areas from inaccessible areas in order to provide access during full power operation as well as D₂O vapor recovery system segregation would have to be designed with ventilation openings in order to prevent buildup of hydrogen following a loss of coolant accident. An alternative solution would be to utilize the vapor recovery systems as part of the safety-related combustible gas control system. As required by GDC 4, internal missile shields would have to be provided around pumps and pieces of redundant equipment. GDC 16 requires that the containment maintain its functional capability following an accident. Since it cannot be demonstrated that an epoxy liner can maintain its integrity following steam jet impingement, it is concluded that leaktightness of the containment would be compromised following such an event. To be licensed in the U.S., the containment would require a suitable metal liner, such as carbon steel.

Discussions related to containment heat removal, cleanup systems, are discussed under that system description.

It appears that GDC 50 is only partially complied with since the containment design bases considers only loss of one-half the primary coolant into the free volume. Since discharge of both loops is a possibility, there is reasonable doubt that all energy sources have been considered. It is unknown whether or not the present design of the HWR ferritic steel forming a part of the containment system is subjected to impact requirements, as related to GDC 51.

It appears that GDC 52 and GDC 53 are met. From the information available, no redundancy exist in the containment ventilation/filtering system as required by GDC 60.

1.3.2.2.4.3 Recommended Design Changes

- GDC 1 - Upgrade QA procedures.
- GDC 2 - Design for negative pressure due to tornado.
- GDC 3 - Provide for atmospheric mixing following LOCA.
- GDC 4 - Design for externally and internally generated missiles.
- GDC 16 - Provide carbon steel liner.
- GDC 39 - Provide inservice inspection program.
- GDC 40 - Refer to Chapter 6 and GDC 32 in Volume I.
- GDC 41 - Provide redundant cleanup system with recirculation capability.
- GDC 42 - Provide inservice inspection program system.
- GDC 43 - Refer to GDC 42 in Volume I.
- GDC 50 - Design for discharge of all primary coolant.
- GDC 51 - Select all materials of penetrations, including that portion of process lines classified as containment, and isolation valves constructed from ferritic materials to meet fracture toughness requirements of GDC 51.
- GDC 52 - None.
- GDC 53 - None.
- GDC 60 - Provide redundant filter system on containment ventilation system.

1.3.2.3 Service Building

1.3.2.3.1 Function

The purpose of the Service Building is to provide a safe enclosure for those systems essential for the safe operation of the plant.

1.3.2.3.2 Description

The Service Building is divided into two main areas called the Reactor Auxiliary Bay and the Service Wing. The Reactor Auxiliary Bay contains those facilities which are closely associated with the actual operation of the Reactor and which can conveniently be located outside of the Reactor Building. The Service Wing houses those facilities of a more general nature and which serve the plant as a whole.

The layout provides for efficient operation of the plant in terms of traffic patterns, radiation zoning, and routing of services between the buildings of the plant. Equipment, instrumentation and piping have been, as far as is practical, located within the Service Building to permit access during plant operation.

The building is a two story or in some areas, a three story steel beam and concrete slab structure with a basement. General arrangement drawings are shown in Figures 1.3-16 through 1.3-22.

An elevator is provided in the Service Wing connecting the first floor, the mezzanine floor and the second floor.

The basement of the Service Building contains the following equipment and systems:

- a) moderator purification,
- b) heavy water storage tanks,
- c) spent fuel bay purification and cooling,
- d) reactor vault purification and cooling system,
- e) liquid radwaste tanks and transfer pumps
- f) emergency core cooling recovery pumps and heat exchanger
- g) off-gas system
- h) secondary control area

The first floor of the Service Wing houses the following facilities:

- a) mechanical workshops,
- b) general maintenance workshop
- c) crane hall
- d) fueling machine decontamination and maintenance rooms
- e) decontamination center
- f) spent fuel storage and handling bays
- g) new fuel storage facility
- h) annulus gas system
- i) D_2O and helium addition stations
- j) radiation protection office and stores
- k) ventilation exhaust room
- l) heavy water vapor recovery system

- m) heavy water management area

In addition, the spent fuel storage area comprises one main bay which provides storage for approximately 8 years and one full core as well as two auxiliary bays.

The mezzanine floor contains workshop offices, storage facilities electrical workshops and cable spreading room.

The second floor of the Service Wing provides room for the following facilities:

- a) personnel lockers and shower facilities
- b) laundry
- c) chemistry and radiochemistry labs and offices
- d) control and instrumentation workshop
- e) control center complex
- f) control room air conditioning system
- g) ventilation intake room
- h) equipment airlock to Reactor Building
- i) resin addition room

1.3.2.3.3 Design Bases

There is insufficient information available to determine the design bases for the Service Building.

1.3.2.3.4 Safety Evaluation

1.3.2.3.4.1 Applicable NRC Criteria

GDC 1 - Quality standards and records.

- GDC 2 - Design Bases for protection against natural phenomena.
- GDC 3 - Fire protection.
- GDC 4 - Environmental and missile design bases.
- GDC 19 - Control room.
- GDC 22 - Protection system independence.
- GDC 23 - Protection system failure modes.
- GDC 24 - Separation of protection and control systems.
- GDC 60 - Control of release of radioactive materials to the environment.
- GDC 61 - Fuel storage and handling and radioactivity control.
- GDC 62 - Prevention of criticality in fuel storage and handling.

1.3.2.3.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with in the design and construction of the containment. The program would have to be expanded to include all phases of construction including the material manufacturer and suppliers.

GDC 2 requires that the building be designed as a Seismic Category I structure since it contains systems important to safety. The present design utilizing steel beam and slab construction with some portions seismically qualified does not comply with GDC 2.

The extent of dousing systems in the Service Building cannot be assessed due to lack of information. However, the area below the control center which houses the cable spreading room makes the plant susceptible to common

mode failure in the event of a fire in the cable spreading area.

The location of the control center on the top floor of the building with a built up steel decking roof makes the control room susceptible to externally generated missiles, which does not meet the requirements of GDC 4. Further, for reasons discussed above, the location of the control room, instrumentation room and cable spreading room would have to be changed in order to meet the requirements of General Design Criteria 22, 23, and 24.

Although the ventilation system in the Service Building draws air from regions of low activity to higher activity, it is passed through a single filter prior to discharge, and the monitoring system is supplied from Class II power. The limits for release generally concur with those of 1CRP. GDC 60 requires that foldup capacity be provided for gaseous and liquid discharges. Further, 10 CFR 50, Appendix I, requires demonstration that added radwaste cleanup be provided unless a cost benefit analysis shows that further additions are not cost effective. Since this additional equipment must fit into the Service Building, the building dimensions will of necessity have to be expanded.

The Service Building contains the spent fuel pool with an epoxy liner, which is not considered a leaktight membrane by U.S. licensing requirements and does not meet the requirements of GDC 61 for fuel storage. Refer also to discussions on spent fuel storage cooling systems for further identification of design changes.

Since the HWR plant uses natural uranium fuel, prevention of criticality in the fuel storage bays is not considered a problem.

1.3.2.3.4.3 Recommended Design Changes

- GDC 1 - Upgrade QA & QC procedures.
- GDC 2 - Design of all Service Building in a Seismic Category I structure.
- GDC 3 - Provide adequate fire protection system in all areas of the Service Building.
- GDC 4 - Design Service Building against missiles, both externally and internally generated.
- GDC 19 - Design control room against missiles, fire and provide with remote air intakes for protection against toxic chemical release.
- GDC 22 - Provide separate cable spreading room for different safety related systems.
- GDC 23 - See discussion on GDC 22.
- GDC 24 - See discussion on GDC 22.
- GDC 60 - Provide liquid and gaseous holdup capacity prior to discharge.
- GDC 61 - Provide steel liner in spent fuel storage bays.
- GDC 62 - None.

1.3.2.4 Turbine Building

1.3.2.4.1 Function

The function of the Turbine Building is to house the turbine generator and its accessories.

1.3.2.4.2 Description

The Turbine Building has a reinforced concrete substructure, steel frame superstructure, steel roof trusses and insulated metal walls and roof. The building contains the turbine generator, condensers, main steam piping and feedwater heaters, feedwater pumps, electrical switchgear, water treatment plant and emergency diesel generators. Turbine Building general arrangements are shown in Figures 1.3-23 through 1.3-28.

The approximate overall dimensions of the building are 360 feet long by 195 feet wide by 110 feet high (above grade). Overhead cranes are provided in the turbine hall for erection and maintenance of the turbogenerator and some of its auxiliaries.

1.3.2.4.3 Design Bases

Insufficient information exist to determine the design bases.

1.3.2.4.4 Safety Evaluation

1.3.2.4.4.1 Applicable NRC Criteria

None.

1.3.2.4.4.2 Assessment

Although the Turbine Building is not a safety related structure, the design reviewed contains emergency equipment, such as the diesel generators and emergency feedwater pumps.

1.3.2.4.4.3 Recommended Design Changes

- o Provide separate Category I structure to house emergency diesel generators and related equipment.

- 1 MAIN STEAM SUPPLY PIPING
- 2 BOILERS
- 3 MAIN PRIMARY SYSTEM PUMPS
- 4 FEEDERS
- 5 CALANDRIA ASSEMBLY
- 6 FUEL CHANNEL ASSEMBLY
- 7 FUELLING MACHINE BRIDGE
- 8 MODERATOR CIRCULATION SYSTEM

Figure 1.3-1
Heavy Water Reactor

Figure 1.3-2
Heavy Water Reactor Simplified Flow Diagram

1 MAIN STEAM SUPPLY PIPING	9 FUELLING MACHINE
2 BOILERS	10 FUELLING MACHINE DOOR
3 MAIN PRIMARY SYSTEM PUMPS	11 CATENARY
4 CALANDRIA ASSEMBLY	12 MODERATOR CIRCULATION SYSTEM
5 FEEDERS	13 PIPE BRIDGE
6 FUEL CHANNEL ASSEMBLY	14 SERVICE BUILDING
7 DOUSING WATER SUPPLY	
8 CRANE RAILS	

Figure 1.3-3
Reactor Building

TYPICAL PLANT LAYOUT

PHWR REACTOR PLANT

United engineers & ARCHITECTS PVT. LTD.

FIGURE 1.3-4

RECOMMENDED
PLOT PLAN

e united engineers & constructors inc.

FIGURE I.3-4a

FIGURE 1.3-5
REACTOR BUILDING
PLAN 1-1

FIGURE 1.3-8
REACTOR BUILDING
PLAN 2-2

FIGURE 1.3-8
REACTOR BUILDING
PLAN 4-4

FIGURE 1.3-10
REACTOR BUILDING
PLAN 6-8

FIGURE 1.3-11
REACTOR BUILDING
PLAN 7-7

FIGURE 1.3-12
REACTOR BUILDING
SECTION 2-2

FIGURE 1.3-13
REACTOR BUILDING
SECTION 2-2

**FIGURE 1.3-14
REACTOR BUILDING
SECTION 1-1**

FIGURE 1.3-15
REACTOR BUILDING
SECTION 1-1

ROOM NO.	DESCRIPTION
SI-001	EMERGENCY FUEL COOLING PUMP PIT
SI-002	DYE STORAGE TANK ROOM
SI-003	RADIATION TANK ROOM
SI-004	STORAGE BAY AND SHIELD COOLING ROOM
SI-005	STAIRS
SI-006	PIPE ACCESS
SI-007	UNDERFLOOR PURIFICATION ROOM
SI-008	STAIRS
SI-009	UNDESIGNATED
SI-010	CORRIDOR
SI-011	EMERGENCY AREA
SI-012	OFF GAS MANAGEMENT
SI-013	UNDESIGNATED
SI-014	SECONDARY CONTROL AREA

FIGURE 1.3-16
SERVICE BUILDING
GENERAL ARRANGEMENTS-BASEMENT

FIGURE 1.3-17
SERVICE BUILDING
GENERAL ARRANGEMENTS—FIRST FLOOR

ROOM NO.	NAME
04-001	CABLE ACCESS
0-002	0-002 SPREADING TOWER
0-003	0-003 SPREADING TOWER
0-004	OFFICE
0-005	OFFICE
0-006	STORAGE
0-007	ELECTRICAL WORKSHOP
0-008	OFFICE
0-009	OFFICE

FIGURE 1.3-18
SERVICE BUILDING
GENERAL ARRANGEMENTS-MEZZANINE

FIGURE 1.3-20
SERVICE BUILDING
GENERAL ARRANGEMENTS—ROOF ELEVATIONS

1.3-32

TURBINE BUILDING LAYOUT
Figure 1.3-23

TURBINE BUILDING PLAN LAYOUT

Figure 1.3-24

1.3-34

TURBINE BUILDING LAYOUT

Figure 1.3-25

1.3-35

SECTION S2

TURBINE BUILDING LAYOUT
Figure 1.3-26

TURBINE BUILDING - SERVICE WING PLAN

Figure 1.3-27

1.3-37

TURBINE BUILDING -
Figure

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CHAPTER 2.0

SITE CHARACTERISTICS

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2.2 <u>Assessment</u>	2.1-1

2.0 SITE CHARACTERISTICS

2.1 Introduction

Our review of criteria for siting of nuclear power plants in Canada identified only the following major areas dealing with safety-related site considerations:

- o Geology and seismology
- o Atmospheric dispersion
- o Population density in the site environs

The criteria for determination of the Design Basis Earthquake (DBE) is defined as the maximum seismic acceleration and frequency for the site defined regional tectonic studies.

2.2 Assessment

Two aspects are addressed in siting nuclear power plants in the USA, that is, safety-related site considerations and environmental-related site considerations.

The safety-related considerations include:

- o Geology and seismology
- o Hydrology
 - Flooding
 - Water Supply
- o Industrial, military and transportation facilities, near the site
- o Atmospheric dispersion
- o Population density in the site environs

In addition, the environment-related considerations include:

- Preservation of important habitats
- Migratory routes of important species
- Entrainment and impingement of aquatic organisms
- Entrapment of aquatic organisms
- Water quality
- Consumptive water use
- Local fogging and icing
- Economic impact of preemptive land use
- Established public amenity areas
- Prospective designated amenity areas
- Public planning
- Visual amenities

Implementation of the investigations required by 10 CFR 100, Appendix A, together with the guidelines provided in USNRC Regulatory Guide 1.70, Section 2.5 adds considerable costs to siting of a nuclear power plant in the USA.

In addition to the geology-seismicity requirements, additional investigations are required in siting for assuring an adequate foundation for the structures. These investigations include such things as soil stability and assurance that the soil is not sensitive to liquefaction during seismic vibration.

For sites located near the coastal areas, additional investigations and studies are required. The first, in line with the geology-seismicity studies, includes the investigation of the potential of the site being

subjected to earthquake generated tsunami. If it is determined that a tsunami wave could impinge on the plant structures then a detailed conservative analysis is required to assess the wave height and means of protecting the structures from the water.

The second consideration for a coastal plant site is the protection of the structures from the effect of wind and flooding due to hurricanes. Both of these investigations add cost in siting a plant near the coast in terms of studies and additional protection that must be provided to mitigate the effects of the ocean forces.

Requirements of limiting the radiation release to the environs are similar for Canadian and USA plants. The low-as-reasonably achievable requirements imposed by 10 CFR 50, Appendix I, where the normal operating releases have been substantially reduced from Canadian practices, has increased the plant cost. In order to comply with these requirements, the utilities have tightened up leakage paths and have provided additional barriers to the release of radioactivity to the environs.

The most concise document describing siting requirements for plants constructed in the USA is the USNRC Regulatory Guide 4.7. This guide is appended herein to provide a basis of comparison with other requirements that may have been imposed by the Canadian government for nuclear power plants.

APPENDIX 2A

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 4.7*

GENERAL SITE SUITABILITY CRITERIA FOR NUCLEAR POWER STATIONS

*This guide was initially published as a draft in September 1974.

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. This guide was revised as a result of substantive comments received from the public and additional staff review.

Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Section.

The guides are issued in the following ten broad divisions:

1. Power Reactors	6. Products
2. Research and Test Reactors	7. Transportation
3. Fuels and Materials Facilities	8. Occupational Health
4. Environmental and Siting	9. Antitrust Review
5. Materials and Plant Protection	10. General

Copies of published guides may be obtained by written request indicating the divisions desired to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Office of Standards Development.

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A. INTRODUCTION

The Energy Reorganization Act of 1974 places on the Nuclear Regulatory Commission (NRC) the responsibility for the licensing and regulation of private nuclear facilities from the standpoint of public health and safety. Paragraphs 100.10(b) and (c) of 10 CFR Part 100, "Reactor Site Criteria," require that the population density, use of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology, be taken into account in determining the acceptability of a site for a nuclear power reactor. Seismic and geologic site criteria for nuclear power plants are provided in Appendix A to 10 CFR Part 100. Appendix A to 10 CFR Part 50 establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants; a number of these criteria are directly related to site characteristics as well as to events and conditions outside the nuclear power unit.

The National Environmental Policy Act of 1969 (NEPA) (83 Stat. 852), implemented by Executive Order 11514 and the Council on Environmental Quality's Guidelines of August 1, 1973 (38 FR 20550), requires that all agencies of the Federal Government prepare detailed environmental statements on proposed major Federal actions which can significantly affect the quality of the human environment. A principal objective of NEPA is to require the Federal agency to consider, in its decision-making process, the environmental impacts of each proposed major action and the available alternative actions.

Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection," of Title 10, Code of Federal Regulations, sets forth the Nuclear Regulatory Commission's policy and procedures for the preparation and processing of environmental impact statements and related documents pursuant to Section 102(2)(C) of the NEPA.

The limitations on the Commission's authority and responsibility pursuant to the NEPA imposed by the Federal Water Pollution Control Act (86 Stat. 916) are addressed in an Interim Policy Statement published in the *Federal Register* on January 29, 1973 (38 FR 2679).

This guide discusses the major site characteristics related to public health and safety and environmental issues which the NRC staff considers in determining the suitability of sites for light-water-cooled (LWR) and high temperature gas-cooled (HTGR) nuclear power stations.^a The guidelines may be used by applicants in identifying suitable candidate sites for nuclear power stations. The

^aFor the purposes of this guide, nuclear power station refers to the nuclear reactor unit(s), nuclear steam supply, electric generating units, auxiliary systems, including the cooling system and structures such as docks that are located on a given site, and any new electrical transmission towers and lines erected in connection with the facilities.

decision that a station may be built on a specific candidate site is based on a detailed evaluation of the proposed site-plant combination and a cost-benefit analysis comparing it with alternative site-plant combinations as discussed in Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations."

Chapter 9 of Regulatory Guide 4.2 discusses the selection of a site from among alternative sites. Although it is recognized that planning methods^b will differ among applicants, Chapter 9 states that the applicant should present its site-plant selection process as the consequence of an analysis of alternatives whose environmental costs and benefits were evaluated and compared and then weighed against those of the proposed facility.

This guide is intended to assist applicants in the initial stage of selecting potential sites for a nuclear power station. Each site that appears to be compatible with the general criteria discussed in this guide will have to be examined in greater detail before it can be considered to be a "candidate" site, i.e., one of the group of sites that are to be considered in selecting a "proposed" or "preferred" site.^c

This guide should be used only in the initial stage of site selection because it does not provide detailed guidance on the various relevant factors and format for ranking the relative suitability or desirability of possible sites. This guide provides a general set of safety and environmental criteria which the NRC staff has found to be valuable in assessing candidate site identification in specific licensing cases.

The information needed to evaluate potential sites at this initial stage of site selection is assumed to be limited to that information which may be obtained from published reports, public records, public and private agencies, and individuals knowledgeable about the locality of a potential site. Although in some cases the applicants may have conducted on-the-spot investigations, it is assumed here that these investigations would be limited to reconnaissance-type surveys at this stage in the site selection process.

The safety issues discussed include geologic/seismic, hydrologic, and atmospheric characteristics of proposed sites; potential effects on the station from accidents associated with nearby industrial, transportation, and

^bSite selection methodologies that have been used by the nuclear power industry are described in "Nuclear Power Plant Siting, A Generalized Process," Atomic Industrial Forum, August 1974, National Environmental Studies Project, R-1578.

^cSee Chapter 9 of Regulatory Guide 4.2 for a discussion of site selection procedures. The "proposed" site submitted by an applicant for a construction permit is that site of a number of "candidate" sites which the applicant prefers and on which the applicant proposes to construct a nuclear power station.

military facilities; and population distribution and densities in the site environs as they relate to protecting the general public from the potential radiation hazards of postulated serious accidents. The environmental issues discussed concern potential impacts from the construction and operation of nuclear power stations on ecological systems, water use, land use, the atmosphere, aesthetics, and socioeconomics.

This guide does not discuss details of the engineering designs required to ensure the compatibility of the nuclear station and the site or the detailed information required for the preparation of the safety analysis and environmental reports. In addition, nuclear power reactor site suitability as it may be affected by the Commission's materials safeguards and plant protection requirements for nuclear power plants is not addressed in this guide.

Guidance concerning the siting of offshore nuclear stations, liquid metal fast breeder reactors (LMFBR), and advanced siting concepts such as underground sites and nuclear energy centers is not included in this guide.

A significant commitment of time and resources may be required to select a suitable site for a nuclear power station, including safety and environmental considerations, and to develop an acceptable design for that site. Site selection involves considerations of public health and safety, engineering and design, economics, institutional requirements, environmental impacts, and other factors. The potential impacts of the construction and operation of nuclear power stations on the physical and biological environment and on social, cultural, and economic features^a are usually similar to the potential impacts of any major industrial facility, but nuclear power stations are unique in the degree to which potential impacts of the environment on their safety must be considered. The safety requirements are primary determinants of the suitability of a site for nuclear power stations, but considerations of environmental impacts and public acceptance of nuclear power stations are also important and need to be evaluated.

In the site selection process, coordination between applicants for nuclear power stations and various Federal, State, and local agencies will be useful in identifying potential problem areas.

Appendices A and B of this guide summarize the important safety-related and environmental considerations

^aBiological and physical environment includes geology, geomorphology, surface and groundwater hydrology, climatology, air quality, limnology, water quality, fisheries, wildlife, and vegetation. Social and cultural features include scenic resources, recreation resources, archeological/historical resources, and community resources including land use patterns. From "Development and the Environment: Legal Reforms to Facilitate Industrial Site Selection," final report by the Committee on Environmental Law, American Bar Association, February 1974.

for assessing the site suitability of nuclear power stations.

B. DISCUSSION

1. Geology/Seismology

Nuclear power stations must be designed to prevent the loss of safety-related functions. Generally, the most restrictive safety-related site characteristics considered in determining the suitability of a site are surface faulting, potential ground motion and foundation conditions^b (including liquefaction, subsidence, and landslide potential), and seismically induced floods. Criteria that describe the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability are provided by Appendix A, "Seismic and Geologic Criteria for Nuclear Power Plants," to 10 CFR Part 100. Safety-related site characteristics are identified in Section 2.5 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," and Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants." In addition to geologic and seismic evaluation for assessing seismically induced flooding potential, Section 2.4 of Regulatory Guide 1.70 and Regulatory Guide 1.59 describe hydrologic criteria including coincident flood events that should be considered.

2. Atmospheric Extremes and Dispersion

The potential effect of natural atmospheric extremes (e.g., tornadoes^c and exceptional icing conditions^d) on the safety-related structures of a nuclear station must be considered. However, the atmospheric extremes that may occur at a site are not normally critical in determining the suitability of a site because safety-related structures, systems, and components can be designed to withstand most atmospheric extremes.

The atmospheric characteristics at a site are an important consideration in evaluating the dispersion of radioactive effluents both from postulated accidents and from routine releases in gaseous effluents.^e In addition to meeting the NRC requirements for the dispersion of

^b"Classification, Engineering Properties and Field Exploration of Soils, Intact Rock and In Situ Masses," WASH-1301, March 1974, outlines some of the procedures used to evaluate site foundation properties.

^cRefer to Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

^dRefer to Section 2.4.7 of Regulatory Guide 1.70.

^eRoutine releases of airborne radioactive material must be kept "as low as practicable." [See 10 CFR Part 20, §20.1(c).] The Commission has published a proposed rule for public comment (40 FR 33029) that substitutes "as low as is reasonably achievable" for the older, less precise term "as low as practicable" where it appears in NRC regulations and regulatory guides.

Section 50.34a of 10 CFR Part 50 sets forth the requirements for design objectives for equipment to control releases of radioactive material in effluents from nuclear power reactors.

(Continued)

airborne radioactive material, the station must meet State and Federal requirements of the Clean Air Amendments of 1970 (PL 91-604). This is unlikely to be an important consideration for nuclear power station siting unless (1) a site is in an area where existing air quality is near or exceeds the limits set under the Clean Air Amendments, (2) there is a potential for interaction of the cooling system plume with a plume containing noxious or toxic substances from a nearby facility, or (3) the auxiliary generators are operating.

The atmospheric data necessary for adequate assessment of the potential dispersion of radioactive material from design basis accidents are described in Regulatory Guide 1.23, "Onsite Meteorological Programs." Models and assumptions used for evaluating the potential radiological consequences of certain postulated accidents are provided in Regulatory Guides 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors;" 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors;" 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors;" 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure;" and 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." However, the atmospheric assumptions in the guides may not be appropriate for sites with unusual atmospheric conditions.

In the evaluation of potential sites, onsite atmospheric reconnaissance can determine if the atmospheric conditions at a site are adequately represented

(Continued)

Section 50.36a further provides that, in order to keep power reactor effluent releases as low as practicable, each license authorizing operation of such a facility will include technical specifications regarding the establishment of effluent control equipment and reporting of actual releases.

Appendix I to 10 CFR Part 50, promulgated May 5, 1975 (40 FR 19439), provides numerical guidance for design objectives and technical specification requirements for limiting conditions of operation for light-water-cooled nuclear power plants.

The following regulatory guides are being prepared to assist in application of the numerical guidance in Appendix I:

1. Calculation of Annual Average Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Implementing Appendix I,
2. Calculations of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWRs),
3. Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors (BWRs), and
4. Methods for Estimating Atmospheric Dispersion of Gaseous Effluents from Routine Releases.

by the available atmospheric data for the area. Canyons or deep valleys frequently have atmospheric variables that are substantially different from those variables measured for the general region. Other topographical features such as hills, mountain ranges, and lake or ocean shorelines can affect the local atmospheric conditions at a site and may cause the dispersion characteristics at the site to be less favorable than those in the general area or region. More stringent design or effluent objectives or a larger exclusion area may be required in such cases.

While it is the concentration of radioactivity in the atmosphere at any distance from the point of release, $x(\text{Ci}/\text{m}^3)$, that must be controlled, the ratio x/Q , where $Q(\text{Ci}/\text{sec})$ is the rate of release of radioactivity from the source, has become a commonly evaluated term because it depends only on atmospheric variables and distance from the source.

If the atmospheric conditions are unfavorable with respect to dispersion characteristics at a proposed site, the exclusion area may have to be unusually large to satisfy the dose criteria of 10 CFR Part 100. If under assumed unfavorable atmospheric conditions (see Regulatory Guides 1.3 and 1.4) the dispersion of radioactivity released following a design basis accident is insufficient at the boundary of the exclusion area (see the following section, "Population Considerations") and the outer boundary of the low population zone, the site would not satisfy the requirements of 10 CFR Part 100. Thus, the design of the station would be required to include appropriate and adequate compensating engineered safety features.

Local fogging and icing can result from plumes discharged into the atmosphere from cooling towers, lakes, canals, or spray ponds, but can generally be acceptably mitigated by station design and operational practices. However, some sites have the potential for severe fogging or icing due to local atmospheric conditions. For example, areas of unusually high moisture content that are protected from large-scale airflow patterns are most likely to experience these conditions. The impacts are generally of greatest potential importance relative to transportation or electrical transmission corridors in the vicinity of a site.

A cooling system designed with special consideration for reducing drift may be required due to the sensitivity of the natural vegetation or the crops in the vicinity of the site to damage from airborne salt particles. The vulnerability of existing industries or other facilities in the vicinity of the site to corrosion by drift from cooling tower or spray system drift should be considered. Not only are the amount, direction, and distance of the drift from the cooling system important, but the salt concentration above the natural background salt deposition at the site is also important in assessing drift effects. None of these considerations are critical in evaluating

the suitability of a site, but they could result in special cooling system design requirements or in the need for a larger site to confine the effects of drift within the site boundary. The environmental effects of salt drift are most severe where saline water or water with high mineral content is used for condenser cooling.

Cooling towers produce cloudlike plumes which vary in size and altitude depending on the atmospheric conditions. The plumes are often a few miles in length before becoming dissipated, but the plumes themselves or their shadows could have aesthetic impacts. Visible plumes emitted from cooling towers in the vicinity of airports could cause a hazard to aviation.

3. Population Considerations

A reactor licensee is required by 10 CFR Part 100 to designate an exclusion area and to have authority to determine all activities within that area, including removal of personnel and property. In selecting a site for a nuclear power station, it is necessary to provide for an exclusion area in which the applicant has such authority. The exclusion area must be of such size that doses to individuals at any point on its boundary for 2 hours immediately following the onset of a postulated fission product release are less than certain prescribed values. Transportation corridors, such as highways, railroads, and waterways, are permitted to traverse the exclusion area provided (1) these are not so close to the facility as to interfere with normal operation of the facility and (2) appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway in the case of emergency to protect the public health and safety.

As set forth in 10 CFR Part 100, a nuclear power station site must have a low population zone (LPZ) immediately surrounding the exclusion area in which the population is (a) sufficiently limited in number and (b) distributed in such a way that there is a reasonable probability that appropriate measures could be taken in their behalf in the event of a serious accident. A proposed site will also have a "population center distance," defined as the distance from the nuclear reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents. The population center distance must be at least one and one-third times the distance to the outer boundary of the LPZ. However, 10 CFR Part 100 requires that the LPZ boundary be sufficiently remote that a release of fission products (calculated as a consequence of a postulated accident) will not result in radiation doses to individuals on the outer boundary of the LPZ greater than certain specified values.

WASH-1235, "The Site Population Factor, A Technique for Consideration of Population in Site Comparison," October 1974, discusses a methodology that is

useful in comparing population distributions at alternative sites.

4. Hydrology

4.1 Flooding

Criteria for evaluation of seismically induced floods are provided in Appendix A to 10 CFR Part 100. Regulatory Guide 1.59 describes an acceptable method of determining the design basis floods for sites along streams or rivers and discusses the phenomena producing comparable design basis floods for coastal, estuary, and Great Lakes sites. The effects of a probable maximum flood (as defined in Regulatory Guide 1.59), seiche, surge, or seismically induced flood such as might be caused by dam failures or tsunami on station safety functions can generally be controlled by engineering design or protection of the safety-related structures, systems, and components which are identified in Regulatory Guide 1.29, "Seismic Design Classification." For some river valleys, flood plains, or areas along coastlines, there may not be sufficient information to make the evaluations needed to satisfy the criteria for seismically induced flooding. In such cases, study of the potential for dam failure, river blockage, or diversion in the river system or distantly and locally generated sea waves may be needed to determine the suitability of a site. In lieu of detailed investigations, Regulatory Guide 1.59 and Section 2.4 of Regulatory Guide 1.70 present acceptable analytical techniques for evaluating seismically induced flooding.

4.2 Water Availability

Nuclear power stations require reliable sources of water for steam condensation, service water, emergency core cooling system, and other functions. In regions where water is in short supply, the recirculation of the hot cooling water through cooling towers, artificial ponds, or impoundments has been practiced.

Essential water requirements for nuclear power plants are that sufficient water be available for cooling during plant operation and normal shutdown, for the ultimate heat sink,^a and for fire protection. The limitations imposed by existing laws or allocation policies govern the use and consumption of cooling water at potential sites^b for normal operation. Regulatory Guide 1.27

^aRegulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," provides guidance on water supply for the ultimate heat sink.

^bTo the extent that site selection is dependent on water diversions for consumptive use, allocation of water supply is a function of state statutory and administrative procedures.

A discussion of the establishment of state regulation of water use is provided in "Industrial Developments and the Environment, Legal Reforms to Improve the Decision-Making Process in Industrial Site Selection," Special Committee on Environmental Law of the American Bar Association, August 1973.

discusses the safety requirements. Consumptive use of water may necessitate an evaluation of existing and future water uses in the area to ensure adequate water supply during droughts both for station operation and other water users (i.e., nuclear power station requirements versus public water supply). Regulatory agencies should be consulted to avoid potential conflicts.

Where required by applicable law, demonstration of a request for certification of the rights to withdraw or consume water and an indication that the request is consistent with appropriate State and regional programs and policies should be provided as part of the application for a construction permit or operating license.

The availability of essential water during periods of low flow or low water level is an important initial consideration for identifying potential sites on rivers, small shallow lakes, or along coastlines. Both the frequency and duration of low flow or low level periods should be determined from the historical record and, if the cooling water is to be drawn from impoundments, from projected operating practices.

4.3 Water Quality

Thermal and chemical effluents discharged to navigable streams are governed by the Federal Water Pollution Control Act (FWPCA, PL 92-500), 40 CFR Part 122, 40 CFR Part 423, and State water quality standards. The applicant should also determine other regulations that are current at the time sites are under consideration. Section 401(a)(1) of the FWPCA requires, in part, that any applicant for an NRC construction permit for a nuclear power station provide to the NRC certification from the State that any discharge will comply with applicable effluent limitations and other water pollution control requirements. In the absence of such certification, no construction permit can be issued by the NRC unless the requirement is waived by the State or the State fails to act within a reasonable period of time. A National Pollution Discharge Elimination System (NPDES) permit to discharge effluents to navigable streams pursuant to Section 402 of the FWPCA may be required for a nuclear power station to operate in compliance with the Act, but is not a prerequisite to an NRC construction permit or operating license.

Evaluations of the dispersion and dilution capabilities and potential contamination pathways of the ground water environment under operating and accident conditions with respect to present and future users are required. Potential radiological and nonradiological contaminants of ground water should be evaluated. The suitability of sites in areas with a complex ground water hydrology or of sites located over aquifers that are or may be used by large populations for domestic or

industrial water supplies or for irrigation water can only be determined after reliable assessments have been made of the potential impacts of the reactor plants on the ground water.

Although management of the quality of surface waters is important, water quality per se is not a determining factor in assessing the suitability of a site since adequate design alternatives can generally be developed to meet the requirements of the Federal Water Pollution Control Act and the Commission's regulations implementing NEPA. However, the environmental characteristics or the complexity of the environment at a site and its vicinity may be such that it would be difficult to obtain or develop sufficient information to establish, in a timely manner, that the potential environmental impacts on water quality would be acceptable. Examples of situations that could pose unusual impact assessment or design problems are areas of existing marginal water quality, small bays, estuaries, stratified waters, and sites that would require intake from and discharge to waters of markedly different quality, such as intake of marine water and discharge to an estuary.

The following are examples of potential environmental effects of station construction and operation that must be assessed: physical and chemical environmental alterations in habitats of important species, including plant-induced rapid changes in environmental conditions; changes in normal current direction or velocity of the cooling water source and receiving water; scouring and siltation resulting from construction and cooling water intake and discharge; alterations resulting from dredging and spoil disposal; and interference with shoreline processes.

5. Ecological Systems and Biota

Areas of great importance to the local aquatic ecosystem may present major difficulties in assessing potential impacts on populations of important species or ecological systems. Such areas include those used for breeding (e.g., nesting and spawning), wintering, and feeding, as well as areas where there may be seasonally high concentrations of individuals of important species.³ Where the ecological sensitivity of a site under consideration cannot be established from existing information, more detailed studies, as discussed in Regulatory

³A species, whether animal or plant, is important (for the purpose of this guide) if a specific causal link can be identified between the nuclear power station and the species and if one or more of the following criteria applies:

- (1) If the species is commercially or recreationally valuable,
- (2) If the species is endangered or threatened,
- (3) If the species affects the well-being of some important species within criteria (1) or (2) or if it is critical to the structure and function of a valuable ecological system or is a biological indicator of radionuclides in the environment.

(Continued)

Guide 4.2, may be necessary. Impacts of station construction^a and operation on the biota and ecological systems may be mitigated by design and operational practices if justifiable relative to costs and benefits. In general, the important considerations in the balancing of costs and benefits are (a) the uniqueness of a habitat or ecological system within the region under consideration and (b) the amount of habitat or ecological system that would be destroyed or disrupted relative to the total amount of the habitat or ecological system present in the region or the vulnerability of the reproductive capacity of important species populations to the effects of construction and operation of the plant and ancillary facilities.

The alteration of one or more of the existing environmental conditions may render a habitat unsuitable as a breeding or nursery area. In some cases, organisms use identical breeding and nursery areas each year; if the characteristics of the areas are changed, breeding success may be substantially reduced or enhanced. Destruction of part or all of a breeding or nursery area may cause population shifts that result in increased competition for the remaining suitable areas. Such population shifts cannot compensate for the reduced size of the breeding or nursery areas if the remaining suitable area is already occupied by the species. Some species will desert a breeding area because of man's activities in the proximity to the area, even in the absence of physical disturbance of the actual breeding area.

Of special concern relative to site selection are those unique or especially rich feeding areas that might be destroyed, degraded, or made inaccessible to important species by station construction or operation. Evaluation of feeding areas in relation to potential construction or operation impacts includes the following considerations: size of the feeding area onsite in relation to the total feeding area offsite, food density, time of use, location in relation to other habitats, topography relative to access routes, and other factors (including man's activities). Site modification may reduce the quality of feeding areas by destruction of a portion of the food base, destruction of cover, or both.

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Endangered and threatened species are defined by PL 93-205, the Endangered Species Act of 1973, as follows: "The term 'endangered species' means any species which is in danger of extinction throughout all or a significant portion of its range other than a species of the Class Insecta determined by the Secretary to constitute a pest whose protection under the provisions of this Act would present an overwhelming and overriding risk to man." "The term 'threatened species' means any species which is likely to become an endangered species within the foreseeable future throughout all or a significant portion of its range." Lists of endangered and threatened species are published periodically in the *Federal Register* by the Secretary of the Interior.

^aA compilation of construction practices is provided in "General Environmental Guidelines for Evaluating and Reporting the Effects of Nuclear Power Plant Site Preparation, Plant and Transmission Facilities Construction," Atomic Industrial Forum, February 1974.

Construction and operation of nuclear power stations can create barriers to migration, occurring mainly in the aquatic environment. Narrow zones of passage for migratory animals in some rivers and estuaries may be restricted or blocked by station operation. Partial or complete blockage of a zone of passage may result from the discharge of heat or chemicals to receiving water bodies or the construction and placement of power station structures in the water body. Strong-swimming aquatic animals often avoid waters of adverse quality, but larval and immature forms are usually moved and dispersed by water currents. It is therefore important in site selection that the routes and times of movement of the immature stages be considered in relation to potential effects.

A detailed assessment of potential impact on the species population would be required for sites where placement of intake or discharge structures would markedly disrupt normal current patterns in migration paths of important species. The potentials for impingement of organisms on cooling water intake structures and entrainment of organisms through the cooling system are determined by a number of variables including site characteristics, intake structure design, and placement of the structures at the site.

Site characteristics should be considered relative to design and placement of cooling system features and the potential of the cooling system to hold fish in an area longer than the normal period of migration or to entrap resident populations in areas where they would be adversely affected, either directly or indirectly, by limited food supply or adverse temperatures. Canals or areas where cooling waters are discharged may induce fish to remain in an unnaturally warmed habitat. The cessation of station operation during winter can be lethal to these fish because of an abrupt drop in water temperature.

6. Land Use and Aesthetics

Many impacts on land use at the site and in the site neighborhood due to construction and operation of the plant, transmission lines, and transportation corridors can be mitigated by appropriate designs and practices. Aesthetic impacts can be reduced by selecting sites where existing topography and forests can be utilized for screening station structures from nearby scenic, historical, or recreational resources. Restoration of natural vegetation, creative landscaping,^b and the integration of structures with the environment can mitigate adverse visual impacts.

Preconstruction archeological excavations can usually reduce losses. Short-term salvage archeology may not be

^bStation protection requirements for nuclear safeguards may influence landscape design and clearing of vegetation.

sufficient if extensive or valuable archeological sites are found on the potential site for a nuclear station. For areas of archeological concern, the Chief Archeologist of the National Park Service is an information source, as are the State Archeologist and the State Liaison Officer responsible for the National Historic Preservation Act activities for a particular state.

Proposed alternative land use may render a site unsuitable for a nuclear power station. For example, lands specified by a community (1) as planned for other uses or (2) as restricted to compatible uses vis-a-vis other lands may be unsuitable. Therefore, official land use plans developed by governments at any level and by regional agencies should be consulted for possible conflicts with power station siting. A list of Federal agencies that have jurisdiction or expertise in land use planning, regulation, or management has been published by the Council on Environmental Quality.^a

Another class of impacts involves the preempting of existing land use at the site itself. For example, nuclear power station siting in areas uniquely suited for growing specialty crops may be considered a type of land conversion involving unacceptable economic dislocation.

Sites adjacent to lands devoted to public use may be considered unsuitable. In particular, the use of some sites or transmission line or transportation corridors close to special areas administered by Federal, State, or local agencies for scenic or recreational use may cause unacceptable impacts regardless of design parameters. Such cases are most apt to arise in areas adjacent to natural-resource oriented areas (e.g., Yellowstone National Park) as opposed to recreation-oriented areas (e.g., Lake Mead National Recreation Area). Some historical and archeological sites may also fall into this category. The acceptability of sites near special areas of public use should be determined by consulting cognizant government agencies.

The following Federal agencies should be consulted for the special areas listed:

a. National Park Service (U.S. Department of the Interior)

National Parks; International Parks; National Memorial Parks; National Battlefields, Battlefield Parks and Battlefield Sites; National Military Parks; Historic Areas and National Historic Sites; National Capital Parks; National Monuments and Cemeteries; National Seashores and Lakeshores; National Rivers and Scenic Riverways; National Recreation Areas; National Scenic Trails and Scientific Reserves; National Parkways

^aSee U.S. Council on Environmental Quality, "Preparation of Environmental Impact Statements: Guidelines," 38 FR 20549, August 1, 1973.

b. National Park Service Preservation Program

National Landmarks Program; Historic American Buildings Survey; National Register of Historic Places; National Historical Landmarks Program; National Park Service Archeological Program

c. Bureau of Sport Fisheries and Wildlife (U.S. Department of Interior)

National Wildlife Refuges

d. Forest Service (U.S. Department of Agriculture)

National Forest Wilderness, Primitive Areas, National Forests.

Individual States and local governments administer parks, recreation areas, and other public use and benefit areas. Information on these areas should be obtained from cognizant State agencies such as State departments of natural resources. (See publications such as the "Conservation Directory 1973: A Listing of Organizations, Agencies and Officials Concerned with Natural Resource Use and Management," published by the National Wildlife Federation for state-by-state references.) The Advisory Council on Historic Preservation or the appropriate State historical society should be contacted for information on historic areas.

It should be recognized that some areas, as yet undesignated, may be unsuitable for siting because of public interest in future dedication to public scenic, recreational, or cultural use. Relatively rare land types such as sand dunes and wetlands are prime candidates for such future designation. However, the acceptability of sites for nuclear power stations at some future time in these areas will depend on the existing impacts from industrial, commercial, and other developments.

7. Industrial, Military, and Transportation Facilities

Potential accidents at present or projected nearby industrial, military, and transportation facilities may affect the safety of a nuclear power station.^a A site should not be selected if, in the event of such an accident, it is not possible to safely shut down a plant at that site or if it is not possible to have nearby facilities alter their mode of operation or incorporate features to reduce to an acceptable level the likelihood and severity of such potential accidents.

In the event of an accident at a nearby industrial facility such as a chemical plant, refinery, mining and quarrying operation, oil or gas well, or gas and petroleum product storage installation, it is possible that

^bSection 2.2 of Regulatory Guide 1.70 lists these safety considerations.

missiles, shock waves, flammable vapor clouds, toxic chemicals, or incendiary fragments may result. These may affect the station itself or the station operators in a way that jeopardizes the safety of the station.

Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," describes assumptions acceptable to the NRC staff for use in assessing the habitability of the control room during and after a postulated external release of hazardous chemicals and describes criteria that are generally acceptable to the staff for the protection of the control room operators.

Nearby military facilities, such as munitions storage areas and ordnance test ranges, may threaten station safety. The acceptability of a site depends on establishing, among other things, that the nuclear power station can be designed so its safety will not be affected by an accident at the military installation. Alternatively, an otherwise unacceptable site may become acceptable if the cognizant military organization agrees to change the installation or mode of operation to reduce the likelihood or severity of potential accidents involving the nuclear station to an acceptable level.

An accident during the transport of hazardous materials (e.g., by air, waterway, railroad, highway, or pipeline) near a nuclear power plant may generate shock waves, missiles, and toxic or corrosive gases which can affect the safe operation of the station. The consequences of the accident will depend on the proximity of the transportation facility to the site, the nature and maximum quantity of the hazardous material per shipment, and the layout of the nuclear station. Unless the station can be designed to operate safely in the event of a postulated accident or an enforceable agreement can be reached to limit the transport of hazardous materials or the transportation link can be relocated, the proposed site may not be acceptable.

Airports are transportation facilities that pose specialized hazards to nearby nuclear power stations. Potential threats to stations from aircraft result from the aircraft itself as a missile and from the secondary effects of a crash, e.g., fire.

8. Socioeconomics

Social and economic issues are important determinants of siting policy. It is difficult both to assess the nature of the impacts involved and to determine value schemes for predicting the level or the acceptability of potential impacts.

The siting, construction, and operation of a nuclear power station may have significant impacts on the socioeconomic structure of a community and may place

severe stresses on the local labor supply, transportation facilities, and community services in general. There may be changes in the tax basis and in community expenditures, and problems may occur in determining equitable levels of compensation for persons relocated as a result of the station siting. It is usually possible to resolve such difficulties by proper coordination with impacted communities; however, some impacts may be locally unacceptable and too costly to avoid by any reasonable program for their mitigation. Evaluation of the suitability of a site should therefore include consideration of purpose and probable adequacy of socioeconomic impact mitigation plans for such economic impacts on any community where local acceptance problems can be reasonably foreseen.

Certain communities in a site neighborhood may be subject to unusual impacts that would be excessively costly to mitigate. Among such communities are towns that possess a notably distinctive cultural character, i.e., towns that have preserved or restored numerous places of historic interest, have specialized in an unusual industry or avocational activity, or have otherwise markedly distinguished themselves from other communities.

9. Noise

Noise levels at nuclear stations occur during both the construction and operation phases and could have unacceptable impacts. Cooling towers, turbines, and transformers contribute to the noise levels during station operation.

C. REGULATORY POSITION

1. Geology/Seismology

Sites that include capable faults, as defined in Appendix A to 10 CFR Part 100, are not suitable for nuclear power stations. The state of the art has not progressed to the point at which it is possible to design a nuclear power station for surface or near-surface displacement with a sufficiently high level of confidence to ensure that the integrity of the safety-related features of the plant will remain intact.

Sites within about 5 miles of a surface capable fault greater than 1000 feet in length are usually not suitable for a nuclear power station. In any case, extensive and detailed geologic and seismic field studies and analyses should be conducted for such a proposed site.

Sites located near geologic structures for which an adequate data base to determine "capability" does not exist at the time of application are likely to be subject to a longer licensing process in view of the need for extensive and detailed geologic and seismic investigations of the site and surrounding region and for the rigorous analyses of the site-plant combination.

Sites with competent bedrock for foundations generally have suitable foundation conditions. In regions where there are few or no such sites, it is prudent to select sites in areas with competent and stable solid soils, such as dense sands and glacial tills. Other materials may also provide satisfactory foundation conditions, but in any case, a detailed geologic and geotechnical investigation will be required to determine static and dynamic engineering properties of the material underlying the site in accordance with Sections IV(a)(4) and V(d) of Appendix A to 10 CFR Part 100.

2. Atmospheric Extremes and Dispersion

As noted in Section B.2 of this guide, site atmospheric conditions are site suitability characteristics principally with respect to the calculation of radiation doses resulting from the release of fission products as a consequence of a postulated accident and the establishment of exclusion area boundary, low population zone boundary, and distance to a population center. Accordingly, the regulatory position on atmospheric dispersion of radiological effluents is incorporated into the following section, "Population Considerations."

Nonradiological atmospheric considerations such as local fogging and icing, cooling tower drift, cooling tower plume lengths and plume interactions between cooling tower plumes, and plumes from nearby industrial facilities should be considered in evaluating the suitability of potential sites.

3. Population Considerations

Areas of low population density are preferred for nuclear power station sites. High population densities projected for any time during the lifetime of a station are considered during both the NRC staff review and the public hearing phases of the licensing process. If the population density at the proposed site is not acceptably low, then the applicant will be required to give special attention to alternative sites with lower population densities.

If the population density, including weighted transient population, projected at the time of initial operation of a nuclear power station exceeds 500 persons per square mile averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), or the projected population density over the lifetime of the facility exceeds 1,000 persons per square mile averaged over any radial distance out to 30 miles, special attention should be given to the consideration of alternative sites with lower population densities.

Transient population should be included for those sites where a significant number of people (other than

those just passing through the area) work, reside part-time, or engage in recreational activities and are not permanent residents of the area. The transient population should be taken into account by weighting the transient population according to the fraction of time the transients are in the area.

Based on past experience, the NRC staff has found that a minimum exclusion distance of 0.4 mile, even with unfavorable design basis atmospheric dispersion characteristics, usually provides assurance that engineered safety features can be designed to bring the calculated dose from a postulated accident within the guidelines of 10 CFR Part 100. If the minimum exclusion distance is less than 0.4 mile, it may be necessary to place special conditions on the station design (e.g., added engineered safety features) before the requirements of 10 CFR Part 100 are met. Also, based on past experience, the staff has found that a distance of 3 miles to the outer boundary of the low population zone is usually adequate.

4. Hydrology

4.1 Flooding

To evaluate sites located in river valleys, on flood plains, or along coastlines where there is a potential for flooding, the site suitability studies described in Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," should be made.

4.2 Water Availability

A highly dependable system of water supply sources must be shown to be available under postulated occurrences of natural and site-related accidental phenomena or combinations of such phenomena as discussed in Regulatory Guide 1.59.

To evaluate the suitability of sites, there should be reasonable assurance that permits for consumptive use of water in the quantities needed for a nuclear power plant of the stated approximate capacity and type of cooling system can be obtained by the applicant from the appropriate State, local, or regional bodies.

4.3 Water Quality

The potential impacts of nuclear power stations on water quality are likely to be acceptable if effluent limitations, water quality criteria for receiving waters, and other requirements promulgated pursuant to the Federal Water Pollution Control Act are applicable and satisfied.

The criteria provided in 10 CFR Parts 20 and 50 will be used by the NRC staff for determining permissible

concentrations of radioactive materials discharged to surface water or to ground water.⁴

Aquifers that are or may be used by large populations for domestic, municipal, industrial, or irrigation water supplies provide potential pathways for the transport of radioactive material to man in the event of an accident. To evaluate the suitability of proposed sites located over such aquifers, detailed studies of factors identified in Section 2.4.13 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," should be completed.

5. Ecological Systems and Biota

The ecological systems and biota at potential sites and their environs should be sufficiently well known to allow reasonably certain predictions that there would be no unacceptable or unnecessary deleterious impacts on populations of important species or on ecological systems with which they are associated from the construction or operation of a nuclear power station at the site.

When early site inspections and evaluations indicate that critical or exceptionally complex ecological systems will have to be studied in detail to determine the appropriate plant designs, proposals to use such sites should be deferred unless sites with less complex characteristics are not available.

It should be determined whether any important species (as defined in Section B.5 of this guide) inhabit or use the proposed site or its environs; and the relative abundance and distribution of their populations should be considered. Potential adverse impacts on important species should be identified and assessed. The relative abundance of individuals of an important species inhabiting a potential site should be compared to available information in the literature concerning the total estimated local population. Any predicted impacts on the species should be evaluated relative to effects on the local population and the total population of the species. The destruction of, or sublethal effects on, a number of individuals which would not adversely affect the reproductive capacity and vitality of a population or the crop of an economically important harvestable population or recreationally important population should generally be acceptable, except in the case of certain endangered species. If there are endangered or threatened species at a site, the potential effects should be evaluated relative to the impact on the local population and the total estimated population over the entire range of the species as noted in the literature.

⁴Appendix 1 to 10 CFR Part 50 provides numerical guidance for design objectives and technical specification requirements for limiting conditions of operation for light-water-cooled nuclear power stations.

It should be determined whether there are any important ecological systems at a site or in its environs. If so, determination should be made as to whether the ecological systems are especially vulnerable to change or if they contain important species habitats, such as breeding areas (e.g., nesting and spawning areas), nursery, feeding, resting, and wintering areas, or other areas of seasonally high concentrations of individuals of important species.

The important considerations in the balancing of costs and benefits include the following: the uniqueness of a habitat or ecological system within the region under consideration, the amount of the habitat or ecological system destroyed or disrupted relative to the total amount in the region, and the vulnerability of the reproductive capacity of important species populations to the effects of construction and operation of the station and ancillary facilities.

If sites contain, are adjacent to, or may impact on important ecological systems or habitats that are unique, limited in extent, or necessary to the productivity of populations of important species (e.g., wetlands and estuaries), they cannot be evaluated as to suitability for a nuclear power station until adequate assessments for the reliable prediction of impacts have been completed and the facility design characteristics that would satisfactorily mitigate the potential ecological impacts have been defined. In areas where reliable and sufficient data are not available, the collection and evaluation of appropriate seasonal data may be required.

Migrations of important species and migration routes that pass through the site or its environs should be identified. Generally, the most critical migratory routes relative to nuclear power station siting are those of aquatic species in water bodies associated with the cooling systems. Site conditions that should be identified and evaluated in assessing potential impacts on important aquatic migratory species include (1) narrow zones of passage, (2) migration periods that are coincident with maximum ambient temperatures, (3) potential for major modification of currents by station structures, (4) potential for increased turbidity during construction, and (5) potential for entrapment, entrainment, or impingement by or in the cooling water system, or blocking of migration by facility structures or effluents.

The potential blockage of movements of important terrestrial animal populations due to the use of the site for a nuclear power station and the availability of alternative routes that would provide for maintenance of the species' breeding population should be assessed.

If justifiable relative to costs and benefits, potential impacts of plant construction and operation on the biota and ecological systems can generally be mitigated by adequate engineering design and site planning and by proper construction and operation practice when there is

adequate information about the vulnerability of the important species and ecological systems.

A summary of environmental considerations, parameters, and regulatory positions for use in evaluating the suitability of sites for nuclear power stations is provided in Appendix B to this guide. A discussion of ecological systems and habitats, the level of detail that should be addressed in the site selection process, and the survey, monitoring, and analytical techniques for assessing impacts on important species and ecological systems will be summarized in subsequent appendices to this guide.

6. Land Use and Aesthetics

Land use plans adopted by Federal, State, regional, or local governmental entities should be examined, and any conflict between these plans and use of a potential site should be resolved by consultation with the appropriate governmental entity.

For potential sites on land devoted to specialty crop production where changes in land use might result in market dislocations, a detailed investigation should be provided to demonstrate that potential problems have been identified and resolved.

The potential aesthetic impact of nuclear power stations at sites near natural-resource oriented public use areas is of particular concern, and evaluation of the suitability of such sites is dependent on consideration of specific station design layout. However, existing aesthetic impacts at potential sites should be taken into account as mitigating any requirements for further special design.

7. Industrial, Military, and Transportation Facilities

Potentially hazardous facilities and activities within 5 miles of a proposed site should be identified. If a preliminary evaluation of potential accidents at these facilities indicates that the potential hazards from shock waves and missiles approach or exceed those of the design basis tornado for the region^a or potential hazards such as flammable vapor clouds, toxic chemicals, or incendiary fragments exist, the suitability of the site should be determined by detailed evaluation of the degree of risk imposed by the potential hazard.

The identification of design basis events resulting from the presence of hazardous materials or activities in the vicinity of a nuclear power station is acceptable if the design basis events include each postulated type of accident for which a realistic estimate of the probability of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines exceeds approximately 10^{-7} per year. Because of the difficulty of assigning precise

^aThe design basis tornado is described in Regulatory Guide 1.76, "Design Basis Floods for Nuclear Power Plants."

numerical values to the probability of occurrence of the types of potential hazards generally considered in determining the acceptability of sites for nuclear stations, judgment must be used as to the acceptability of the overall risk presented by an event.

In view of the low probability events under consideration, the probability of occurrence of the initiating events leading to potential consequences in excess of 10 CFR Part 100 exposure guidelines should be based on assumptions that are as realistic as is practicable. In addition, because of the low probability events under consideration, valid statistical data are often not available to permit accurate quantitative calculation of probabilities. Accordingly, a conservative calculation showing that the probability of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines is approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.

The effects of design basis events have been appropriately considered if analyses of the effects of those accidents on the safety-related features of the proposed nuclear station have been performed and appropriate measures (e.g., hardening, fire protection) to mitigate the consequences of such events have been taken.

To evaluate the suitability of sites in detail for potential accidents involving hazardous materials and activities at nearby industrial, military, and transportation facilities, the studies described in Section 2.2 of Regulatory Guide 1.70 should be made.

8. Socioeconomics

The NRC staff considers that an evaluation of the suitability of nuclear power station sites near distinctive communities should demonstrate that the construction and operation of the nuclear station, including transmission and transportation corridors, and potential problems relating to community services, such as schools, police and fire protection, water and sewage, and health facilities, will not adversely affect the distinctive character of the community. A preliminary investigation should be made to identify and analyze problems that may arise due to the proximity of a distinctive community to a proposed site.

9. Noise

Noise levels at proposed sites must comply with applicable Federal, State, and local noise regulations.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Since this guide reflects current NRC staff practice with regard to the implementation of existing regulations concerning site suitability, it can be used immedi-

ately to indicate considerations that should be addressed in the initial stage of the site selection process to identify potential sites for nuclear power stations.

APPENDIX A

SAFETY-RELATED SITE CONSIDERATIONS FOR ASSESSING SITE SUITABILITY FOR NUCLEAR POWER STATIONS

This appendix provides a checklist of safety-related site characteristics, relevant regulations and regulatory guides, and regulatory experience and position for assessing site suitability for nuclear power stations.

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
<p>A.1 Geology/Seismology</p> <p>Geologic and seismic characteristics of a site, such as surface faulting, ground motion, and foundation conditions (including liquefaction, subsidence, and landslide potential), may affect the safety of a nuclear power station.</p>	<p>10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."</p> <p>Regulatory Guide 1.70, Chapter 2 (identifies safety-related site characteristics).</p> <p>Regulatory Guide 1.29 (discusses plant safety features which should be controlled by engineering design).</p>	<p>Sites that include capable faults are not suitable for a nuclear power station.</p> <p>Sites within about 5 miles of a surface capable fault (greater than 1000 feet in length) are generally not suitable for a nuclear power station.</p> <p>Sites should be selected in areas for which an adequate geologic data base exists to determine "capability." Delay in licensing can result from a need for extensive geologic and seismic investigations. Conservative design of safety-related structures will be required when geologic, seismic, and foundation information is questionable.</p> <p>Sites with competent bedrock generally have suitable foundation conditions.</p> <p>If bedrock sites are not available, it is prudent to select sites in areas known to have a low subsidence and liquefaction potential. Investigations will be required to determine the static and dynamic engineering properties of the material underlying the site as stated in 10 CFR Part 100, Sec. IV(a)(4) and Sec. V(d) of Appendix A.</p>

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
<p>A.2 Atmospheric Dispersion</p> <p>The atmospheric conditions at a site should provide sufficient dispersion of radioactive materials released during a postulated accident to reduce the radiation exposures of individuals at the exclusion area and low population zone boundaries to the values prescribed in 10 CFR Part 100.</p>	<p>10 CFR Part 100, "Reactor Site Criteria."</p> <p>Regulatory Guide 1.23, "Onsite Meteorological Programs."</p> <p>Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors."</p> <p>Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."</p> <p>Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors."</p> <p>Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure."</p> <p>Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."</p>	<p>Unfavorable safety-related design basis atmospheric dispersion characteristics can be compensated for by an adequate exclusion distance and engineered safety features. Accordingly, the regulatory position on atmospheric dispersion of radiological effluents is incorporated into the section "Population Considerations" (see A.3 of this appendix).</p>

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
<p>A.3 Population Considerations</p> <p>In the event of a serious accident at a nuclear power station, effective action must be taken to minimize exposure of individuals outside the station to any radioactive materials which may be released during the accident. To ensure that exposure to populations will be minimized in the event of an accident, the nuclear power station should not be located in a densely populated area.</p>	<p>10 CFR Part 100, "Reactor Site Criteria," requires the following:</p> <ul style="list-style-type: none"> • An "exclusion area" surrounding the reactor in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property; • A "low population zone" (LPZ) which immediately surrounds the exclusion area in which the population number and distribution is such that "there is a reasonable probability that appropriate measures could be taken in their behalf in the event of a serious accident;" • At any point on the exclusion area boundary and on the outer boundary of the LPZ the exposure of individuals to a postulated release of fission products (as a consequence of an accident) be less than certain prescribed values, • That the "population center distance," defined as the distance from the nuclear reactor to the nearest boundary of a densely populated center having more than 25,000 residents, be at least one and one-third the distance from the reactor to the outer boundary of the LPZ. <p>Regulatory Guides 1.3, 1.4, 1.5, 1.24, and 1.25 give calculational methods (see A.2 of this appendix.)</p>	<p>If the population density, including weighted transient population, projected at the time of initial operation of a nuclear power station exceeds 500 persons per square mile averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), or the projected population density over the lifetime of the facility exceeds 1,000 persons per square mile averaged over any radial distance out to 30 miles, special attention should be given to the consideration of alternative sites with the lower population densities.</p> <p>Transient population should be included for those sites where a significant number of people (other than those just passing through the area) work, reside part-time, or engage in recreational activities, and are not permanent residents of the area. The transient population should be taken into account by weighting the transient population according to the fraction of time the transients are in the area.</p> <p>Based on past experience, the NRC staff has found that a minimum exclusion distance of 0.4 mile,^a even with the most unfavorable design basis atmospheric dispersion characteristics, provides assurance that engineered safety features can be added that will bring the calculated doses from a postulated accident within the guidelines of 10 CFR Part 100. If the minimum exclusion distance is less than 0.4 mile, it may be necessary to place special conditions on station design (e.g., added engineered safety features) before the site can be considered acceptable. Also based on past experience, the NRC staff has found that a distance of 3 miles to the outer boundary of the LPZ is usually adequate.^a</p>

^aThe guidelines numbers for exclusion area and LPZ are based on historical siting experience of light-water-cooled reactors. In certain instances different dimensions have been established for high temperature gas-cooled reactors.

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
<p>A.4 Hydrology</p> <p>A.4.1 Flooding</p> <p>Precipitation, wind, or seismically induced flooding (e.g., resulting from dam failure, from river blockage or diversion, or from distantly and locally generated sea waves) can affect the safety of a nuclear power station.</p> <p>A.4.2 Water Supply</p> <p>A safety-related water supply is required for normal or emergency shutdown and cooldown.</p> <p>A.4.3 Water Quality</p> <p>Contamination of ground water and surface water by radioactive materials discharged from nuclear stations could cause public health hazards.</p>	<p>10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."</p> <p>Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."</p> <p>Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," (Section 2.4).</p> <p>10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants;" Criterion 2, "Design Bases for Protection Against Natural Phenomena."</p> <p>10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."</p> <p>Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."</p> <p>Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants."</p> <p>10 CFR Part 20, "Standards For Protection Against Radiation."</p> <p>10 CFR Part 50, "Licensing of Production and Utilization Facilities."</p>	<p>To evaluate sites located in river valleys, on flood plains, or along coastlines where there is a potential for flooding, the studies described in Regulatory Guide 1.59 should be made.</p> <p>A highly dependable system of water supply sources should be shown to be available under postulated occurrences of natural phenomena and site-related accidental phenomena or combinations of such phenomena as discussed in Regulatory Guide 1.59.</p> <p>To evaluate the suitability of a site, there must be reasonable assurance that permits for water use and for water consumption in the quantities needed for a nuclear power plant of the stated approximate capacity and type of cooling system can be obtained by the applicant from the appropriate State, local, or regional bodies.</p> <p>The criteria provided in 10 CFR Parts 20 and 50 will be used by the NRC staff for determining permissible concentrations of radionuclides discharged to surface water and ground water.</p>

Considerations	Relevant Regulations and Regulatory Guides	Regulatory Experience and Position
<p>A.5 Industrial, Military, and Transportation Facilities Near the Site</p> <p>Accidents at present or projected nearby industrial, military, and transportation facilities may affect the safety of the nuclear power station.</p>	<p>10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 4, "Environmental and Missile Design Bases."</p> <p>Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports," Section 2.2 (lists types of facilities and potential accidents).</p> <p>Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."</p>	<p>Potentially hazardous facilities and activities within 5 miles of a proposed site must be identified. If a preliminary evaluation of potential accidents of these facilities indicates that the potential hazards from shock waves and missiles approach or exceed those of the design basis tornado for the region (the design basis tornado is described in Regulatory Guide 1.76), or potential hazards such as flammable vapor clouds, toxic chemicals, or incendiary fragments exist, the suitability of the site should be determined by detailed evaluation of the potential hazard.</p> <p>The identification of design basis events resulting from the presence of nearby hazardous materials or activities in the vicinity of a nuclear power station is acceptable if the design basis events include each postulated type of accident for which a realistic estimate of the probability of occurrence of potential exposures in excess of 10 CFR Part 100 guidelines exceeds approximately 10^{-7} per year.</p> <p>To evaluate the suitability of sites in detail for potential accident situations involving hazardous materials and activities from nearby industrial, military, and transportation facilities, the studies described in Section 2.2 of Regulatory Guide 1.70 should be made.</p>

APPENDIX B

ENVIRONMENTAL CONSIDERATIONS FOR ASSESSING SITE SUITABILITY FOR NUCLEAR POWER STATIONS

This appendix summarizes environmental considerations related to site characteristics that should be addressed in the early site selection process. The relative importance of the different factors to be considered varies with the region or State in which the potential sites are located.

Site selection processes can be facilitated by establishing limits for various parameters based on the best judgment of specialists knowledgeable of the region under consideration. For example, limits can be chosen for the

fraction of water that can be diverted in certain situations without adversely affecting the local populations of important species. Although simplistic because important factors such as the distribution of important species in the water body are not taken into account, such limits can be useful in a screening process for site selection.

A discussion of performance characteristics of light-water-cooled reactor stations which may affect the environment is given in WASH-1355, "Nuclear Power Facility Performance Characteristics for Making Environmental Impact Assessments," December 1974.

Considerations	Parameters	Regulatory Position
<p>B.1 Preservation of Important Habitats</p> <p>Important habitats are those that are essential to maintaining the reproductive capacity and vitality of important species populations^a or the harvestable crop of economically or recreationally important species. Such habitats include breeding areas (e.g., nesting and spawning areas), nursery, feeding, resting, and wintering areas or other areas of seasonally high concentrations of individuals of important species.</p> <p>The construction and operation of nuclear power stations (including new transmission lines and access corridors constructed in conjunction with the station) can result in the destruction or alteration of habitats of important species leading to changes in the abundance of a species or in the species composition of a community.</p>	<p>The proportion of an important habitat that would be destroyed or significantly altered in relation to the total habitat within the region in which the proposed site is to be located is a useful parameter for estimating potential impacts of the construction or operation of a nuclear power station. The value of the proportion varies among species and among habitats. The region considered in determining proportions is the normal geographic range of the specific population in question.</p> <p>If endangered or threatened species occur at a site, the potential effects of the construction and operation of a nuclear power station should be evaluated relative to the potential impact on the local population and the total estimated population over the entire range of the species.</p> <p>See also Chapter 2 of Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations."</p>	<p>In general, a detailed justification should be provided when the destruction or significant alteration of more than a few percent of important habitat types is proposed.</p> <p>The reproductive capacity of populations of important species and the harvestable crop of economically or recreationally important populations must be maintained unless justification for proposed or probable changes can be provided.</p>

^aAs defined for this guide in Section B.

Considerations	Parameters	Regulatory Position
<p>B.2 Migratory Routes of Important Species</p> <p>Seasonal or daily migrations are essential to maintaining the reproductive capacity of some important species populations.</p> <p>Disruption of migratory patterns can result from partial or complete blockage of migratory routes by structures, discharge plumes, environmental alterations, or man's activities (e.g., transportation or transmission corridor clearing and site preparation).</p>	<p>The width or cross-sectional area of a water body at a proposed site relative to the general width or cross-sectional area in the portion of the water body used by migrating species should be estimated.</p> <p>Suggested minimum zones of passage range from 1/3 to 3/4 of the width or cross-sectional areas of narrow water bodies.^{a, b}</p> <p>Some species migrate in central, deeper areas while others use marginal, shallow areas. Rivers, streams, and estuaries are seldom homogeneous in their lateral dimension with respect to depth, current velocity, and habitat type. Thus, the use of width or cross-sectional area criteria for determining adequate zones of passage should be combined with a knowledge of important species and their migratory requirements.</p>	<p>Narrow reaches of water bodies should usually be avoided as sites for locating intake or discharge structures.</p> <p>A zone of passage that will permit normal movement of important species populations and maintenance of the harvestable crop of economically important populations should be provided.</p>

^a *Water Quality Criteria*, 1972, National Academy of Sciences-National Academy of Engineering, Washington, D.C., 1972.

^b *Handbook of Environmental Control, Volume III: Water Supply and Treatment*, R.G. Bond and C.P. Straub (Editors), CRS Press, Cleveland, Ohio, 1973.

Considerations	Parameters	Regulatory Position
<p>B.3 Entrainment and Impingement of Aquatic Organisms</p> <p>Plankton, including eggs, larvae, and juvenile fish, can be killed or injured by entrainment through power station cooling systems or in discharge plumes.</p> <p>The reproductive capacity of important species populations may be impaired by lethal stresses or by sublethal stresses that affect reproduction of individuals or result in increased predation on the affected species population.</p> <p>Fish and other aquatic organisms can be killed or injured by impingement on cooling water intake screens^a or by entrainment in discharge plumes.</p>	<p>The depth of the water body at the point of intake relative to the general depth of the water body in the vicinity of the site.</p> <p>The proportion of water withdrawn relative to the net new available water at the site is an indirect measure of the destruction of plankton which in turn is indicative of possible effects on populations of important species. It has been suggested that the fraction of available new water that can be diverted is in the range of 10% to 20% of flow.^{b,c}</p> <p>This simplistic parameter (proportion of water withdrawal) is suitable for use in a screening process for site selection. However, other factors such as distribution of important species should be considered and in all cases the advice of experts on the local fisheries should be consulted to ensure that proposed withdrawals will not be excessive.</p>	<p>The site should have characteristics that allow placement of intake structures where the relative abundance of important species is small and where low approach velocities can be attained. (Deep regions are generally less productive than shallow areas. It is not implied that benthic intakes are necessary.)</p> <p>Important habitats (see B.1) should be avoided as locations for intake structures.</p>

^aApproach velocity and screen-face velocity are design criteria that may affect the impingement of larger organisms, principally fish, on intake screens. Acceptable approach and screen-face velocities are based on fish swim speeds which will vary with the species, site, and season.

^b*The Water's Edge: Critical Problems of the Coastal Zone*, B.H. Ketchum (Editor), MIT Press, Cambridge, Mass., 1972.

^c*Engineering for Resolution of the Energy-Environment Dilemma*, National Academy of Engineering, Washington, D.C., 1972.

Considerations	Parameters	Regulatory Position
<p>B.4 Entrapment of Aquatic Organisms</p> <p>Cooling water intake and discharge system features, such as canals and thermal plumes, can attract and entrap organisms, principally fish. The resulting concentration of important fish species near the station site can result in higher mortalities from station-related causes, such as impingement, cold shock, or gas bubble disease, than would otherwise occur.</p> <p>Entrapment can also interrupt normal migratory patterns.</p>	<p>Site characteristics that will accommodate design features that mitigate or prevent entrapment.</p>	<p>Sites where the construction of intake or discharge canals would be necessary should be avoided unless the site and important species characteristics are such that entry of important species to the canal can be prevented or limited by screening.</p>

Considerations	Parameters	Regulatory Position
<p>B.5 Water Quality</p> <p>Effluents discharged from nuclear power plants are governed under the authority of the Federal Water Pollution Control Act (FWPCA)—(PL 92-500).</p>	<p>Applicable EPA-approved State water quality standards.</p> <p>For states without EPA-approved water quality standards, the water quality criteria listed in <i>Water Quality Criteria, 1972</i>^a will be used for evaluation.</p>	<p>Pursuant to Section 401(a)(1) of the FWPCA, certification from the State that any discharge will comply with applicable effluent limitations and other water pollution control requirements is necessary before the NRC can issue a construction permit unless the requirement is waived by the State or the State fails to act within a reasonable length of time.</p> <p>Issuance of a permit pursuant to Section 402 of the Act is not a prerequisite to an NRC license or permit.</p> <p>Where station construction or operation has the potential to degrade water quality to the possible detriment of other users, more detailed analyses and evaluation of water quality may be necessary.</p>

^a*Water Quality Criteria, 1972*, National Academy of Sciences—National Academy of Engineering, Washington, D.C. 1972.

Considerations	Parameters	Regulatory Position
<p>B.6 Water Availability</p> <p>The consumptive use of water for cooling may be restricted by statute, may be inconsistent with water use planning, or may lead to an unacceptable impact to the water resource.</p>	<p>Applicable Federal, State, and local statutory requirements.</p> <p>Compatibility with water use plan of cognizant water resource planning agency.</p> <p>In the absence of a water use plan, the effect on other water users is evaluated considering flow or volume reduction and the resultant ability of all users to obtain adequate supply and to meet applicable water quality standards (see B.5, Water Quality).</p>	<p>Water use and consumption must comply with statutory requirements and be compatible with water use plans of cognizant water resources planning agencies.</p> <p>Consumptive use should be restricted such that the supply of other users is not impaired and that applicable surface water quality standards could be met, assuming normal station operational discharges and extreme low flow conditions defined by generally accepted engineering practices.</p> <p>For multipurpose impounded lakes and reservoirs, consumptive use should be restricted such that the magnitude and frequency of draw-down will not result in unacceptable damage to important habitats (see B.1, Preservation of Important Habitats) or be inconsistent with the management goals for the water body.</p>

Considerations	Parameters	Regulatory Position
<p>B.7 Established Public Amenity Areas</p> <p>Areas dedicated by Federal, State, or local governments to scenic, recreational, or cultural purposes are generally prohibited areas for siting power stations.</p> <p>Siting nuclear power stations in the vicinity of established public amenity areas could result in the loss or deterioration of important public amenities.</p>	<p>Proximity to public amenity area. Viewability (see B.10, Visual Amenities).</p>	<p>Siting in the vicinity of designated public amenity areas will generally require extensive evaluation and justification.</p> <p>The evaluation of the suitability of sites in the vicinity of public amenity areas is dependent on consideration of a specific plant design and station layout in relation to potential impacts on the public amenity area.</p>
<p>B.8 Prospective Designated Amenity Areas</p> <p>Areas containing important resources for scenic, recreational, or cultural use may not currently be designated as such by public agencies but may involve a net loss to the public if converted to power generation. These areas may include locally rare land types, such as sand dunes, wetlands, or coastal cliffs.</p>	<p>Comparison of possible amenity areas in number and extent with other similar areas available on a local, regional, or national basis, as appropriate.</p>	<p>Public amenity areas that are distinctive, unique, or rare in a region should be avoided as sites for nuclear power stations.</p>

Considerations	Parameters	Regulatory Position
<p>B.9 Public Planning</p> <p>Land use for a nuclear power station should be compatible with established land use or zoning plans of governmental entities.</p>	<p>Officially adopted land use plans.</p>	<p>Land use plans adopted by Federal, State, regional, or local government entities must be examined, and any conflict between these plans and use of a proposed site must be resolved by consultation with the appropriate governmental entity.</p>
<p>B.10 Visual Amenities</p> <p>The presence of power station structures may introduce adverse visual impacts to residential, recreational, scenic, or cultural areas or other areas with significant dependence on desirable viewing characteristics.</p>	<p>The solid angle subtended by station structures at critical viewing points.</p>	<p>The visual intrusion of nuclear power station structures as viewed from nearby residential, recreational, scenic, or cultural areas should be controlled by selecting sites where existing topography and forests can be utilized for screening station structures from those areas in which visual impacts would otherwise be unacceptable.</p>

Considerations	Parameters	Regulatory Position
<p>B.11 Local Fogging and Icing</p> <p>Water and water vapor released to the atmosphere from recirculating cooling systems can lead to ground fog and ice resulting in transportation hazards and damage to electric transmission systems.</p>	<p>Increase in number of hours of fogging or icing caused by operation of the station.</p>	<p>The hazards on transportation routes from fog or ice that result from station operation should be evaluated. The evaluation should include estimates of frequency of occurrence of station-induced fogging and icing and their impact on transportation, electrical transmission, and other activities and functions.</p>
<p>B.12 Cooling Tower Drift</p> <p>Concentrations of chemicals, dissolved solids, and suspended solids in cooling tower drift could affect terrestrial biota and result in unacceptable damage to vegetation and other resources.</p>	<p>The percent drift loss from recirculating condenser cooling water, particle size distribution, salt deposition rate, local atmospheric conditions, and loss of sensitive terrestrial biota affected by salt deposition from cooling tower drift.</p>	<p>The potential loss of important terrestrial species and other resources should be considered.</p>

Considerations	Parameters	Regulatory Position
<p>B.13 Cooling Tower Plume Lengths</p> <p>Natural draft cooling towers produce cloudlike plumes which vary in size and altitude depending on the atmospheric conditions. The plumes are usually a few miles in length before becoming dissipated, although plume lengths of 20 to 30 miles have been reported from cooling towers. Visible plumes emitted from cooling towers could cause a hazard to commercial and military aviation in the vicinity of commercial and military airports. The plumes themselves or their shadows could have aesthetic impacts.</p> <p>B.14 Plume Interaction</p> <p>Water vapor from cooling tower plumes may interact with industrial emissions from nearby facilities to form noxious or toxic substances which could cause adverse public health impacts, or result in unacceptable levels of damage to biota, structures, and other resources.</p>	<p>The number of hours per year the plume is visible as a function of direction and distance from the cooling towers.</p> <p>The degree to which impacts may occur will vary depending on the distance between the nuclear and fossil-fueled sites, the hours per year of plume interaction, the type and concentration of chemical reaction products, the area of chemical fallout, and the local atmospheric conditions.</p>	<p>The visibility of cooling tower plumes as a function of direction and distance from cooling towers should be considered. The evaluation should include estimates of frequency of occurrence for plumes as well as potential hazards to aviation in the vicinity of commercial and military airports.</p> <p>The hazards to public health, structures, and other resources from potential plume interaction between cooling tower plumes and plumes from fossil-fueled sites and industrial emissions from nearby facilities should be considered.</p>

Considerations	Parameters	Regulatory Position
<p>B.15 Noise</p> <p>Undesirable noise levels at nuclear power stations could occur during both the construction and operation phases and have unacceptable impacts near the plant.</p>	<p>Applicable Federal, State, and local noise regulations.</p>	<p>Noise levels at proposed sites must comply with statutory requirements.</p>
<p>B.16 Economic Impact of Pre-emptive Land Use</p> <p>Nuclear power stations can pre-empt large land areas, especially when large cooling lakes are constructed. The land requirement is likely to be an important issue when a proposed site is on productive land (e.g., agricultural land) that is locally limited in availability and is important to the local economy, or which may be needed to meet foreseeable national demands for agricultural products.</p>	<p>The level of local economic dislocation, such as loss of income, jobs, and production, caused by pre-emptive use of productive land and its effect on meeting foreseeable national demands for agriculture products.</p>	<p>If a preliminary evaluation of net local economic impact of the use of productive land for a nuclear power station indicates a potential for large economic dislocation, the NRC staff will require a detailed evaluation of the potential impact and justification for the use of the site based on a cost-effectiveness comparison of alternative station designs and site-station combinations. To complete its evaluation, the staff will also need information on whether and to what extent the land use affects national requirements for agricultural products.</p>

INDEX

CHAPTER 3.0

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

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3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance to NRC General Design Criteria

A discussion of the HWR compliance to the General Design Criteria of 10 CFR 50, Appendix A, is given in Volume 1.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classifications

Structures, systems, and components important to safety are to be designed to remain functional in the event of a Safe Shutdown Earthquake (SSE) and are classified as seismic Category I. These plant features are also referred to as safety-related in this report. These include, but are not limited to those structures, systems, and components which are necessary.

- o To assure the integrity of the Reactor Coolant Pressure Boundary (RCPB);
- o To shutdown the reactor and maintain it in a safe shutdown condition; and
- o To prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposure of 10 CFR 100.

Seismic Category I structures, systems, and components are to be designed to remain functional, as required, to safely shutdown the reactor and maintain it in a safe condition after a SSE event.

All structures surrounding seismic Category I systems and components also are designed for Category I. Subsystems which are not Category I are to be isolable and physically separated from Category I subsystems to preclude affecting the performance of Category I system safety-related functions by the failure of non-Category I systems or components.

A comparison of seismic Category I structures show the following:

STRUCTURE REQUIREMENTS

<u>Structures</u>	<u>USA Requirements</u>	<u>HWR Requirements</u>
1. Containment	Seismic Category I	Seismic Category I
2. Reactor Service Bldg.	Seismic Category I	Note 1
3. Control Building	Seismic Category I	Note 1
4. Diesel Generator Bldg.	Seismic Category I	Note 1
5. Diesel Fuel Storage Tank Foundation	Seismic Category I	Not Identified

NOTE 1:

The design description of these structures do not identify that they meet seismic Category I requirements imposed in the USA. These requirements, in addition to seismic requirements, includes tornado loadings and missile protection.

3.2.2 QUALITY GROUP CLASSIFICATION

The three-level "defense-in-depth" safety concept which has evolved in and now characterizes the U.S. nuclear power program is the basic consideration in assigning Quality Group classifications to systems and components.

At the first level, assurance is provided that the design is safe in normal operation and has a maximum tolerance for errors, abnormal operation, and component malfunction.

At the second level, protection is provided against incidents which might be anticipated to occur despite the care taken in design, construction, and operation to prevent them. This additional level of protection for the operating personnel and the public is provided by reliable protection devices and systems, designed to assure that such incidents will be prevented, arrested, or accommodated safely.

Finally, at the third level public safety is further assured by evaluating the margins of plant safety by testing the design against the calculated consequences of hypothetical events. These evaluations permit judgments to be made regarding the desirability of providing additional safety margins. These safety considerations are applied in such a way that safety is an integral part of all phases of the program.

This "defense-in-depth" concept is achieved through provision of multiple barriers that must be breached or penetrated before the radioactive material in the fuel could be released to the environs. The first barrier is the fuel and its metal cladding which is designed to retain fission product gases. The second independent barrier is the Reactor Coolant

Pressure Boundary (RCPB) system. The third barrier is the Containment system which includes an essentially leak-tight conservatively designed structure having a series of automatically actuated valves to isolate the containment atmosphere from the environs and the public.

Quality Group, or Safety Classification is a continuance of this "defense-in-depth" philosophy. The Quality Group classification is based on the relative importance of a system or component to function in such a manner to provide safe operation of the plant. Classification is further assigned to those systems and components so as to assure their ability to perform their designated safety-function during postulated improbable events to safely shut down the reactor and maintain it in that condition.

The highest level of quality is assigned to the Reactor Coolant Pressure Boundary system. This is self-evident since it is the system which must be designed to operate continuously in a safe and reliable manner. The components in the RCPB system must be designed and evaluated to accommodate the many normal and abnormal transients postulated to occur during the plant life.

The next level of quality is assigned to the Engineered Safeguard Features (ESF) of the plant. The ESF systems are provided to mitigate the effects of postulated events of the RCPB. The ESF systems are required to function, and must be designed to function, under abnormal loading conditions.

There are other systems in the plant that provide auxiliary functions in the normal plant operation. In addition, there are systems provided for the handling and storage of fuel, and for treating radioactive waste. These systems should be designed to function, as designed, under normal and upset plant operating conditions. In the event of abnormal plant operating conditions, the systems and components should be evaluated to ensure that they retain their pressure boundary integrity and prevent the release of radioactivity to the environs.

The Quality Group concept, as related to Safety Class, provides a gradation in the Quality level commensurate with the requirements of the system and component function. These requirements then require that uniform criteria is applied to all components in the same Quality Group. The criteria includes all the requirements for the construction of the component and system.

A review of the classification criteria applied to the HWR's and to the LWR's (as constructed in the USA) show variations, as shown in Table 3.2-1. These differences would have to be resolved prior to licensing such a plant in the USA.

TABLE 3.2-1

QUALITY GROUP CLASSIFICATION

COMPARISONUSA RequirementsQuality Group A

All those Pressure-Containing Components of Boiling and Pressurized Water-Cooled Nuclear Power Reactors, such as Pressure Vessels, Piping, Pumps, and Valves, which are:

1. Part of the Reactor Coolant System, or
2. Connected to the Reactor Coolant System, up to and including any and all of the following:
 - a. The outermost containment isolation valve in System Piping which penetrates Primary Reactor Containment.
 - b. The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment.
 - c. The Reactor Coolant System Safety and Relief Valves.

Components which are connected to the Reactor Coolant System and are part of the Reactor Coolant Pressure Boundary defined above shall be classified other than Safety Class 1, provided:

HWR RequirementsGroup 1 - Preventative

- 1(a). Systems whose failure could cause a radioactive release to the public in the absence of further protective action (e.g. Heat Transport System inside containment).

TABLE 3.2-1 (cont'd)

USA Requirements

Quality Group A (cont'd)

1. In the event of Postulated failure of the Component during normal reactor operation, the Reactor can be shutdown and cooled down in an orderly manner, assuming makeup is provided by the Reactor Coolant Makeup System Only, or
2. The Component is or can be isolated from the Reactor Coolant System by two valves (Both Closed, Both Open, or One Closed and The Other Open). Each open valve must be capable of Automatic Actuation and, assuming the other valve is open. It's closure time must be such that, in the event of Postulated failure of the Component during normal Reactor Operation, each valve remains operable and the Reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the Reactor Coolant Makeup System Only.

Quality Group B

Quality Group B Standards should be applied to Water-And-Steam-containing Pressure Vessels (other than Turbines and Condensers). Storage Tanks, Piping, Pumps, and Valves, that are a part of the Reactor Coolant Boundary but are excluded from the Requirements of Quality Group A, or not a part of the Reactor Coolant Pressure Boundary but part of:

HWR Requirements

Group 1 - Preventative (cont'd)

Group 1 - Preventative

- 1(c). Systems required to shutdown the reactor in a normal manner, and to maintain it in a safe shutdown condition by removing the decay heat, thereby avoiding the hazard of a subsequent process failure as discussed in 1(a). above (e.g., control absorbers, boilers.)

TABLE 3.2-1 (cont'd)

USA Requirements

Quality Group B (cont'd)

1. Systems or Portions of Systems* that are required for:
 - a. Emergency Core Cooling
 - b. Post-Accident Containment Heat Removal
 - c. Post-Accident Containment Atmosphere Cleanup
2. Systems or Portions of Systems* that are required for:
 - a. Reactor Shutdown
 - b. Residual Heat Removal
3. Those Portions of the Steam Systems of Boiling Water Reactors extending from the Outermost Containment Isolation Valve up to but not including the Turbine Stop Valves, and connected Piping up to and including the First Valve that is either normally closed or capable of Automatic Closure during all modes of Normal Reactor Operation.
4. Those portions of the Steam and Feedwater Systems of Pressurized Water Reactors extending from and including the Secondary side of Steam Generators up to and including the Outermost Containment Isolation

HWR Requirements

Group 1 - Preventative (cont'd)

Group 2 - Protective

- 2(b). Those systems designed to remove decay heat and limit the radioactive release subsequent to the failure discussed in 1(a) above.
- 2(a). Those systems designed to quickly shutdown the nuclear reaction in the event of the failure discussed in 1(a) above.

NOTE: *The System Boundary includes those Portions of the System required to accomplish the Specified Safety Function and Connected Piping Up To and Including the First Valve (including a Safety or Relief Valve) that is either normally closed or capable of Automatic Closure when the Safety Function is Required.

TABLE 3.2-1 (cont'd)

USA Requirements

Quality Group B (cont'd)

Valves, and connected piping up to and including the First Valve (including a Safety or Relief Valve) that is either normally closed or capable of Automatic Closure during all modes of Normal Reactor Operation.

5. Systems or Portions of Systems that are connected to the Reactor Coolant Pressure Boundary and are not capable of being isolated from the Boundary during all modes of Normal Reactor Operation by the two valves, each of which is either normally closed or capable of Automatic Closure.

Quality Group C

Quality Group C Standards should be applied to Water, Steam, and Radioactive Waste containing Pressure Vessels (other than Turbines and Condensers), Storage Tanks, Piping, Pumps, and Valves not part of the Reactor Coolant Pressure Boundary nor included in Quality Group B Systems but part of:

HWR Requirements

Group 1 - Preventative

- 1(b). Systems or features designed without further protective system lines of defense, whose failure could cause a radioactive release to the public (e.g., spent fuel bay outside of containment).

TABLE 3.2-1 (cont'd)

USA Requirements

Quality Group C (cont'd)

1. Cooling Water and Auxiliary Feedwater Systems or Portions of these Systems** that are required for:
 - a. Emergency Core Cooling
 - b. Post-Accident Containment Heat Removal
 - c. Post-Accident Containment Atmosphere Cleanup
 - d. Residual Heat Removal from the Reactor and from the Spent Fuel Storage Pool.

Portions of these Systems required for their Safety Functions that do not operate during any mode of Normal Reactor Operation or can not be tested adequately should be classified as Quality Group B.

2. Cooling water and seal water systems or portions of these Systems** that are required for functioning of Reactor Coolant System Components important to Safety, such as Reactor Coolant Pumps.
3. Systems or Portions of Systems that are connected to the Reactor Coolant Pressure Boundary and are capable of being isolated from that Boundary during all modes of Normal Reactor Operation by two valves, each of which is normally closed or capable of Automatic Closure.**

NOTE: **Components in influent lines may be Classified as Quality Group D provided they are capable of being isolated from the Reactor Coolant Pressure Boundary by an additional valve which has High Leak-Tight Integrity.

HWR Requirements

Group 1 - Preventative (cont'd)

- 1(d). Those parts of the plant whose failure during certain defined incidents could indirectly cause a radioactive release by adversely affecting the operation of the systems defined above, (e.g., a structure collapsing during an earthquake).

Group 2 - Protective

- 2(c). Those systems required to support proper operation of protective systems defined in 2(a) and 2(b) above.

TABLE 3.2-1 (cont'd)

USA Requirements

HWR Requirements

Quality Group C (cont'd)

4. Radioactive Waste Treatment, Handling and Disposal Systems, and other Systems that contain or may contain Radioactive Material. Components in these Systems (Except those included in 1., 2., and 3. above) whose failure would not result in Potential Exposures in excess of 0.5 REM Whole Body (or its equivalent to parts of the body) at the Site Boundary or beyond may be classified as Quality Group D.

Quality Group D

Quality Group D should be applied to Water-And Steam-Containing Components not part of the Reactor Coolant Pressure Boundary nor classed as Quality Group B or C but part of Systems or Portions of Systems that contain or may contain Radioactive Material.

In addition to Quality Group Classification of systems and components of a nuclear power plant, additional requirements must be imposed on systems and components to identify their Conditions of Design.

The Conditions of Design represent the combination of the plant process operating conditions and the natural phenomena which must be accommodated in the plant. The natural phenomena includes earthquakes, wind, flooding, and tornadoes. These combinations can impose large loadings on the systems and components.

To properly identify how a component should perform under various plant operation Conditions of Design, it is necessary to identify all of the sources of loading and the stress level to accommodate the loads. The ASME Boiler & Pressure Vessel Code provides design rules and criteria for the design and construction of components but does not provide guidance for selection of the proper criteria.

It must be recognized that there is a difference between the Conditions of Design for a Plant Operating Condition, the Component Operating Condition, and the System Operating Condition to accommodate the Plant Condition.

An example of these differences can best be shown as:

Consider a plant undergoing an Emergency Plant Condition imposing high loadings on the components and initiating the controls of an Engineered Safeguards System (ESF System). This condition will cause high stresses in the Reactor Coolant Pressure Boundary (RCPB) components. Since the ESF System is provided to operate under abnormal conditions to provide safeguards to the plant, the abnormal loads must be considered as normal loads

to the system. These abnormal loads being considered as normal loads is important to components that must perform a motion in order to perform their safeguards function (such as a pump or valve). These "active" components should be designed to accommodate the abnormal loads as normal design stress levels.

Passive components, such as piping systems, could be designed to accommodate abnormal loads at a somewhat higher stress level provided there is assurance that the pipe system and supports will stay intact and will not crimp in such a manner as to block or restrict flow.

Conditions of Design is another restriction imposed on the plant designer to more fully describe how the components are to function. The designer must first classify the component, describe the Conditions of Design for the various loads imposed on the system or component, and to describe the horizontal and vertical seismic accelerations that the component will be subjected to under the SSE or OBE.

The present definitions for Plant Conditions of Design (COD) vary between the various standards and guides. A definition recently proposed by the ANS is:

Normal Plant Condition (COD I): Normal Plant Condition defines the plant safety status during events which are planned to occur regularly in the course of plant operation. These events include: startup, power operation, shutdown, refueling, and routine testing and maintenance of components and systems.

Upset Plant Condition (COD II): Upset Plant Condition defines the plant safety status following unplanned events which are expected to occur occasionally or with moderate frequency during the plant life. These events could be normal operation plus the OBE seismic event.

Emergency Plant Condition (COD III): Emergency Plant Condition defines the plant safety status following events which may occur infrequently during the plant life, such as a SSE or loss of offsite power.

Faulted Plant Conditions (COD IV): Faulted Plant Conditions defines the plant safety status following events which are limiting faults and are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactivity and because they represent upper bounds on failures or accidents with a probability of occurrence sufficiently high to require consideration in the design.

Table 3.2.2 provides safety criteria associated with the various Plant Conditions of Design.

Table 3.2-3 provides a listing of the safety classifications of the various systems and components of a HWR nuclear power plant. Codes, seismic category, and appropriate QA programs are listed where the information was available. In addition, this same type information is shown for "Recommended USA Classification".

TABLE 3.2.2

PLANT SAFETY CRITERIA
AND
PLANT CONDITIONS OF DESIGN

Plant Condition	Criteria
Normal (COD I)	<ul style="list-style-type: none"> a. The plant shall be designed to accommodate Normal Plant Conditions such that releases of radioactive material from the plant are within the limits of Title 10 CFR 20 and Appendix I of Title 10 CFR 50. b. Normal Plant Conditions shall be accommodated within the designated code and standard requirements. c. Normal Plant Conditions shall be accommodated without the need for action by the plant protection system. d. By itself, a Normal Plant Condition shall not generate a more serious plant condition without other incidents or failures occurring independently. e. Normal Plant Conditions shall not cause a subsequent loss of function of the primary coolant boundary or the containment.
Upset (COD II)	<ul style="list-style-type: none"> a. The plant shall be designed to accommodate Upset Plant Conditions such that releases of radioactive material from the plant do not result in whole body doses to any member of the general public in excess of 500 mrem/yr. b. Upset Plant Conditions shall be accommodated within the designated code and standard requirements. c. Upset Plant Conditions shall be accommodated with, at most, an orderly shutdown of the reactor and with the plant capable of returning to operation without extensive corrective action or repair.

TABLE 3.2.2 (cont'd)

Plant Condition	Criteria
Emergency (COD III)	<p>d. By itself, an Upset Plant Condition shall not generate a more serious plant condition without other incidents or failures occurring independently.</p> <p>e. Upset Plant Conditions shall not cause a subsequent loss of function of the primary coolant boundary or the containment.</p>
	<p>a. The plant shall be designed to accommodate Emergency Plant Conditions such that releases of radioactive material from the plant do not result in whole body doses to any member of the general public in excess of:</p> <p>(1) 500 mrem/yr. for events anticipated to occur during the plant life or</p> <p>(2) the 10 CFR 100 limits for events not anticipated to occur during the plant life.</p>
	<p>b. Emergency Plant Conditions shall be accommodated within the designated code and standard requirements.</p>
	<p>c. Emergency Plant Conditions shall be accommodated with a safe shutdown of the reactor and with the plant capable of returning to operation following corrective action or repair of damage.</p>
	<p>d. By itself, an Emergency Plant Condition shall not generate a more serious plant condition without other incidents or failures occurring independently.</p> <p>e. Emergency Plant Conditions shall not cause a subsequent loss of function of the primary coolant boundary or the containment.</p>

TABLE 3.2.2 (cont'd)

Plant Condition	Criteria
Faulted (COD IV)	<ul style="list-style-type: none"> a. The plant shall be designed to accommodate Faulted Plant Conditions such that releases of radioactive material from the plant are within the limits of 10 CFR 100. b. Faulted Plant Conditions shall be accommodated within the designated code and standard requirements. c. Faulted Plant Conditions shall be accommodated with a safe shutdown and cooldown of the reactor. Sufficient damage may occur so as to preclude resumption of operation.

The assignment of Faulted Plant Condition is based on the frequency of the event being extremely low. Such events are not anticipated to occur over the life of the plant, but are postulated to establish a basis for the safety design of the plant. Designers may choose to make the consequences of a Faulted Plant Condition repairable, rather than a plant "write-off", to provide more assurance of preserving the capital investment of the plant.

TABLE 3.2-3

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS										
System or Component	HWR Classification					Recommended USA Classification				
	Safety Class	ASME Code	Seismic Category	QA Program	Note	Safety Class	ASME Code	Seismic Category	QA Program	Note
Reactor	Note 1	Note 2	Note 3	Note 4		Note 1	Note 2	Note 3	Note 5	
Calandria-End Shield Assembly									App. B	
o Calandria	3	III-1	I	QA-1		1	III-1	I	App. B	
o Calandria Tubes	3	III-1	I	QA-2		1	III-1	I	App. B	
o End Shields & Supports	NA	VIII-2	I	QA-1		1	III-1	I	App. B	1/2
o Shield Cooling System	NA	VIII-2	I	-		2	III-1	I	App. B	1/2
Calandria Vault	-	-	-	-		1	III-2	I	App. B	8
Reactivity Control Units										
o In-Core Flux Detectors	-	-	-	-		NA	NA	I	App. B	
o Ion-Chamber Units	-	-	-	-		NA	NA	I	App. B	
o Adjuster Units	-	-	-	-		2	NA	I	App. B	
o Solid Control Absorbers	2	NA	I	QA-2		2	NA	I	App. B	
o Liquid Zone Control	3	III-1	I	QA-2		3	III-1	I	App. B	
o Shutoff Unit (Shutdown #1)	3	III-1	I	QA-2		3	III-1	I	App. B	
o Liquid Inject. (Shutdown #2)	1	III-1	I	QA-1		1	III-1	I	App. B	
o Viewing Ports	-	-	-	-				I	App. B	
o Reactivity Mechanism Deck	-	-	-	-		1	III-1	I	App. B	
PRIMARY HEAT TRANSPORT AND ASSOCIATED SYSTEMS										
Primary Heat Transport System										
o Reactor Fuel Channel Assemblies	1	III-1	I	QA-1		1	III-1	I	App. B	
o Reactor Coolant Piping	1	III-1	I	-		1	III-1	I	App. B	
o Steam Generators	1/2	III-1	I	QA-1		1/2	III-1	I	App. B	
o Pressurizer	1	III-1	I	-		1	III-1	I	App. B	
o Over Pressure Protection	1	III-1	I	-		1	III-1	I	App. B	
o Heat Transport Pumps	1	III-1	I	QA-1		1	III-1	I	App. B	

TABLE 3.2-3 (cont'd)

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS											
System Component	HWR Classification					Recommended USA Classification					
	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note	
Primary Heat Transport Purification Circuit											
o Filters	1	III-1	-	QA-3		1	III-1	I	App. B		
o Ion Exchange Columns	1	III-1	-	QA-2		1	III-1	I	App. B		
o Interchanger/Cooler	1	III-1	-	QA-2		1	III-1	I	App. B		
o Piping Valves	1	III-1	-	QA-3		1	III-1	I	App. B		
Gland Seal System											
o Coolers	1/3	III-1	I	QA-1		1/3	III-1	I	App. B		
o Filters	1/3	III-1	I	QA-2		1/3	III-1	I	App. B		
o Piping, Valves	1/3	III-1	I	QA-2		1/3	III-1	I	App. B		
Shutdown Cooling System											
o Heat Exchangers	1/NN	III-1/VIII	I	QA-1		1	III-1	I	App. B		
o Pumps	1	III-1	I	QA-2		1	III-1	I	App. B		
o Piping, Valves	1	III-1	I	QA-2		1	III-1	I	App. B		
Pressure & Inventory Control											
o Pressurizer	1	III-1	I	-		1	III-1	I	App. B		
o Piping, Valves	1	III-1	I	-		1	III-1	I	App. B		
o Heat Exchanger & Condenser	1/NN	III-1/VIII	I	-		1/3	III-1	I	App. B		
o Pressure Relief System	1	IV-1	I	-		1	III-1	I	App. B		
o Storage Tank	1	III-1	I	-		1	III-1	I	App. B		
D ₂ O Collection System											
o Collection Tank	3	III-1	I	-		3	III-1	I	App. B		
o Pumps	3	III-1	I	-		3	III-1	I	App. B		
o Piping Valves	3	III-1	I	-		3	III-1	I	App. B		

TABLE 3.2-3 (cont'd)

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS											
System Component	HWR Classification					Recommended USA Classification					
	Safety Class	ASME Code	Seismic Category	QA Program	Note	Safety Class	ASME Code	Seismic Category	QA Program	Note	
D ₂ O Sampling System											
o Sample Coolers	3	III-1	I	-		3	III-1	I	App. B		
o Filters	3	III-1	I	-		3	III-1	I	App. B		
o Piping, Valves	3	III-1	I	-		3	III-1	I	App. B		
Steam and Feedwater											
o Steam System											
o Piping, Valves	2/NN	III-1/B31.1	-	-	6	2/NN	III-1/B31.1	I	App. B	7	
o Safety & Relief	2	III-1	-	-	-	2	III-1	I	App. B		
o Feedwater System											
o Piping, Valves	2	III-1	-	-	6	2/NN	III-1/B31.1	I	App. B	7	
SAFETY SYSTEMS											
Containment Systems											
o Containment											
o Structure			I			2	III-2	I	App. B		
o Air Locks	2	III/MC	I	QA-2		2	III-MC	I	App. B		
o Penetrations	2	III/MC	I	QA-2		2	III-MC	I	App. B		
o Lining	-	-	-			Steel Liner	III-2	I	App. B		
o Heat Removal Systems											
o Dousing System											
o Piping, Valves	2	III-1	I	QA-2		2	III-1	I	App. B		
o Water Cooler	2/NN	III-1/VII	I	-		2/3	III-1	I	App. B		
o Pumps	2	III-1	I	-		2	III-1	I	App. B		
o Building Air Coolers											
o Blowers	-	-	-	-		2	III-1	I	App. B		
o Cooling Coils	-	-	-	-		2	III-1	I	App. B		
o Piping, Valves	-	-	-	-		2	III-1	I	App. B		

TABLE 3.2-3 (cont'd)

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS										
System Component	HWR Classification					Recommended USA Classification				
	Safety Class	ASME Code	Seismic Category	QA Program	Note	Safety Class	ASME Code	Seismic Category	QA Program	Note
o Containment Isolation										
o Ventilation Ducts	2	III-1	I	-		2	III-1	I	App. B	
o Isolation Valves	2	III-1	I	-		2	III-1	I	App. B	
o Supporting Systems										
o Vapor Recovery	2	III-1	I	-		2	III-1	I	App. B	
o Reactor Bldg. Vent.	2	III-1	I	-		2	III-1	I	App. B	
Emergency Core Cooling										
o Recovery Pumps	2	III-1	I	QA-2		2	III-1	I	App. B	
o Heat Exchangers	2	III-1	I	QA-2		2	III-1	I	App. B	
o Piping, Valves	2	III-1	I	QA-2		2	III-1	I	App. B	
Shutdown System No. 1 (Activates Absorber Elements)	3	III-1	I	-				I	App. B	
Shutdown System No. 2										
o Helium Pressure Vessel	3	III-1	I	-		3	III-1	I	App. B	
o Piping, Valves	1/3	III-1	I	-		1/3	III-1	I	App. B	
o Poison Tanks	1	III-1	I	-		1	III-1	I	App. B	
Emergency Water Supply										
o Diesel Driven Pumps	-	-	-	-		3	III-1	I	App. B	
o Diesel Fuel Tanks	-	-	-	-		3	III-1	I	App. B	
o Diesel Starting System	-	-	-	-		3	III-1	I	App. B	
o Piping, Valves	-	-	-	-		3	III-1	I	App. B	
Onsite Power System										
o Standby AC Power										
o Diesel Generators	-	-	-	-		NN	NA	Non-I	NA	
o Emergency Power Supply										
o Diesel Generators	-	-	-	-		B	NA	I	App. B	

TABLE 3.2-3 (cont'd)

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS										
System Component	HWR Classification					Recommended USA Classification				
	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note
<u>Auxiliary and Service Systems</u>	:	:	:	:	:	:	:	:	:	:
Spent Fuel Handling and Storage	:	:	:	:	:	:	:	:	:	:
o Spent Fuel Discharge Pool	3	-	I			3	ACI	I	App. B	
o Spent Fuel-Reception Bay Pool	3	-	I			3	ACI	I	App. B	
o Spent Fuel-Defective Fuel Bay Pool	3	-	I			3	ACI	I	App. B	
o Spent Fuel-Storage Bay Pool	3	-	I			3	ACI	I	App. B	
o Spent Fuel Discharge and Transfer Equipment	3	NA	I	QA-2/3		3	III-1	I	App. B	
o Cranes & Hoists										
o Spent Fuel Bay Cooling and Purification										
o Pumps	3	III-1	I	QA-2		3	III-1	I	App. B	
o Heat Exchangers	3/NN	III-1/VIII	I	QA-2		3	III-1	I	App. B	
o Ion Exchange Columns	3	III-1	I	QA-3		3	III-1	I	App. B	
o Piping, Valves	3	III-1	I	QA-3		3	III-1	I	App. B	
Fuel Handling System										
o Fueling Machines	1	III-1	I	QA-1/2		1	III-1	I	App. B	
o Bridge Cranes	1	NA	I	QA-3		2	NA	I	App. B	
o D ₂ O System	2	III-1	I	QA-2		2	III-1	I	App. B	
o Compressed Air System	NA	B31.1	-	QA-2						
o Oil Hydraulic System	NA	B31.1	-	QA-2						
o Piping, Valves	2	III-1	I	QA-2		2	III-1	I	App. B	

TABLE 3.2-3 (cont'd)

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS										
System Component	HWR Classification					Recommended USA Classification				
	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note
Cooling Water Systems	:	:	:	:	:	:	:	:	:	:
o Raw Service Water	:	:	:	:	:	:	:	:	:	:
o Pumps	-	-	-	-	:	NN	VIII-1	Non-I	NA	:
o Piping, Valves	-	-	-	-	:	NN	B31.1	Non-I	NA	:
o Strainers	-	-	-	-	:	NN		Non-I	NA	:
o Recirculated Cooling Water	:	:	:	:	:	:	:	:	:	:
o Heat Exchangers	-	-	-	-	:	3	III-1	I	App. B	10
o Pumps	-	-	-	-	:	3	III-1	I	App. B	10
o Piping, Valves	-	-	-	-	:	3	III-1	I	App. B	10
Demineralized Makeup Water	:	:	:	:	:	:	:	:	:	:
o Reactivator	-	-	-	-	:	3	III-1	I	App. B	:
o Filters	-	-	-	-	:	3	III-1	I	App. B	:
o Ion Exchangers	-	-	-	-	:	3	III-1	I	App. B	:
o Storage Tanks	-	-	-	-	:	3	III-1	I	App. B	:
o Pumps	-	-	-	-	:	3	III-1	I	App. B	:
o Piping, Valves	-	-	-	-	:	3	III-1	I	App. B	:
Chilled Water System	:	:	:	:	:	:	:	:	:	:
o Refrigeration System	NN	-	NA	-	:	3	III-1	I	App. B	:
o Heat Exchangers	NN	-	NA	-	:	3	III-1	I	App. B	:
o Pumps	NN	-	NA	-	:	3	III-1	I	App. B	:
o Piping, Valves	NN	B31.1	NA	-	:	3	III-1	I	App. B	:
Process Auxiliary Systems	:	:	:	:	:	:	:	:	:	:
o Compressed Air	:	:	:	:	:	:	:	:	:	:
o Compressors	NN	-	NA	-	:	NN	NA	Non-I	NA	:
o Coolers	NN	-	NA	-	:	NN	VIII-1	Non-I	NA	:
o Separators	NN	-	NA	-	:	NN	VIII-1	Non-I	NA	:
o Air Receivers	NN	-	NA	-	:	3	III-1	I	App. B	:
o Air Dryers	NN	-	NA	-	:	3	III-1	I	App. B	:
o Air Filters	NN	-	NA	-	:	3	III-1	I	App. B	:
o Piping, Valves	NN	B31.1	NA	-	:	3	III-1	I	App. B	:

TABLE 3.2-3 (cont'd)

System Component	SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS									
	HWR Classification					Recommended USA Classification				
	Safety Class	ASME Code	Seismic Category	QA Program	Note	Safety Class	ASME Code	Seismic Category	QA Program	Note
o Bottled Gas Systems										
o CO ₂ Piping, Valves	3	III-1	I	-		3	III-1	I	App. B	
o H ₂ Piping, Valves	1/3	III-1	I	-		1/3	III-1	I	App. B	
o N ₂ Piping, Valves	NN	B31.1	Non-I	-		NN	B31.1	Non-I	NA	
o He Piping, Valves	1	III-1	I	-		1	III-1	I	App. B	
o D ₂ O Sampling Systems										
o Sample Coolers	3	III-1	I	-		3	III-1	I	App. B	
o Filters	3	III-1	I	-		3	III-1	I	App. B	
o Sample Cannisters	3	III-1	I	-		3	III-1	I	App. B	
o Piping, Valves	3	III-1	I	-		3	III-1	I	App. B	
o Failed Fuel Detection										
o Piping, Valves	-	-	-	-		3	III-1	I	App. B	
o Annulus Gas System										
o Piping, Valves	3	III-1	I	-		3	III-1	I	App. 3	
o Shield Cooling System										
o Pumps	NN	-	-	-		3	III-1	I	App. 3	
o Heat Exchangers	NN	VIII	-	-		3	III-1	I	App. 3	
o Ion Exchangers	NN	VIII	-	-		3	III-1	I	App. 3	
o Tanks	NN	VIII	-	-		3	III-1	I	App. 3	
o Piping, Valves	NN	B31.1	-	-		3	III-1	I	App. 3	
o Resin Transfer System	3/NN	-	-	-						
o Tank	NN	VIII	-	-	9					
o Piping, Valves	3/NN	III-1/B31.1	I	-	9	3	III-1	I	App. B	
						3	III-1	I	App. B	

TABLE 3.2-3 (cont'd)

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS											
System Component	HWR Classification					Recommended USA Classification					Note
	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note	
o Deuteration and De- Deuteration											
o Resins Tanks	3	III-1	I	-		3	III-1	I	App. B		
o Ion Exchangers	3	III-1	I	-		3	III-1	I	App. B		
o Resin Disposal Tanks	3	III-1	I	-		3	III-1	I	App. B		
o Pump	3	III-1	I	-		3	III-1	I	App. B		
o D ₂ O and H ₂ O Tanks	3	III-1	I	-		3	III-1	I	App. B		
o Piping, Valves	3	III-1	I	-		3	III-1	I	App. B		
HVAC Systems											
o Reactor Bldg.	-	-	-	-		2	III-1	I	App. B		
o Service Bldg.	-	-	-	-		3	III-1	I	App. B		
o Turbine Bldg.	-	-	-	-		NN	NA	Non-I	NA		
o Turbine Auxiliary Bay and Service Wing	-	-	-	-		3	III-1	I	App. B		
o Battery Room	-	-	-	-		NN	NA	Non-I	NA		
o Central Heating Plant	-	-	-	-		NN	NA	Non-I	NA		
o Diesel Rooms	-	-	-	-		NN	NA	Non-I	NA		
o Administration Bldg.	-	-	-	-		NN	NA	Non-I	NA		
Moderator and Auxiliary											
o Main Moderator System											
o Heat Exchangers	3	III-1/VIII:	I	-		3	III-1	I	App. B		
o Pumps	3	III-1	I	-		3	III-1	I	App. B		
o Piping, Valves	3	III-1	I	-		3	III-1	I	App. B		
o Head Tank	3	III-1	I	-		3	III-1	I	App. B		

TABLE 3.2-3 (cont'd)

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS										
System Component	HWR Classification					Recommended USA Classification				
	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note
o Moderator Auxiliary Systems:										
o Purification System										
o Ion Exchangers	3	III-1	I	-		3	III-1	I	App. B	
o Heat Exchanger	3	III-1	I	-		3	III-1	I	App. B	
o Filter	3	III-1	I	-		3	III-1	I	App. B	
o Strainer	3	III-1	I	-		3	III-1	I	App. B	
o Piping, Valves	3	III-1	I	-		3	III-1	I	App. B	
o Moderator Cover Gas System										
o Compressors	3	III-1	I	-		3	III-1	I	App. B	
o Recombiners	3	III-1	I	-		3	III-1	I	App. B	
o Gas Cooler	3	III-1	I	-		3	III-1	I	App. B	
o Piping, Valves	3	III-1	I	-		3	III-1	I	App. B	
o Liquid Poison System										
o Mixing Tanks	3	III-1	I	-		3	III-1	I	App. B	
o Agitators	3	NA	I	-		3	NA	I	App. B	
o Piping, Valves	1/3	-	-	-		1/3	III-1	I		
o Poison Tanks	1	III-1	I	-		1	III-1	I	App. B	
o D ₂ O Collection System										
o Collection Tank	3	III-1	I	-		3	III-1	I	App. B	
o Transfer Pump	3	III-1	I	-		3	III-1	I	App. B	
o Piping, Valves	3	III-1	I	-		3	III-1	I	App. B	
Heavy Water Management										
o D ₂ O Supply System										
o Storage Tanks	3	III-1	I	-		3	III-1	I	App. B	
o Pumps	3	III-1	I	-		3	III-1	I	App. B	
o Piping, Valves	3	III-1	I	-		3	III-1	I	App. B	

TABLE 3.2-3 (cont'd)

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS											
System Component	HWR Classification					Recommended USA Classification					
	Safety	ASME	Seismic	QA	Note	Safety	ASME	Seismic	QA	Note	
	Class	Code	Category	Program		Class	Code	Category	Program		
o D ₂ O Vapor Recovery	:	:	:	:	:	:	:	:	:	:	:
o Air Dryers	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Duct Work	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Pumps	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Collection Tanks	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Piping, Valves	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o D ₂ O Cleanup System	:	:	:	:	:	:	:	:	:	:	:
o Charcoal Filters	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Ion Exchangers	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Pumps	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Piping, Valves	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o D ₂ O Upgrading System	:	:	:	:	:	:	:	:	:	:	:
o Rectification Column	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Reboiler	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Condenser	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Cold Trap	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Evaporator	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Pumps	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Tanks	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
o Piping, Valves	: 3	: III-1	: I	: -	:	: 3	: III-1	: I	: App. B	:	:
Other Auxiliary Systems	:	:	:	:	:	:	:	:	:	:	:
o Fire Protection	: NN	: NFPA	: NA	: -	:	: NN	: NFPA	: -	: -	:	:
o D-G Fuel Oil System	:	:	:	:	:	:	:	:	:	:	:
o Storage Tanks	: -	: -	: -	: -	:	: 3	: Note	: I	: App. B	:	:
o Transfer Pumps	: -	: -	: -	: -	:	: 3	: Note	: I	: App. B	:	:
o Piping, Valves	: -	: -	: -	: -	:	: 3	: Note	: I	: App. B	:	:

TABLE 3.2-3 (cont'd)

SYSTEM AND COMPONENT SAFETY CLASSIFICATIONS										
System Component	HWR Classification					Recommended USA Classification				
	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note	Safety : Class	ASME : Code	Seismic : Category	QA : Program	Note
<u>TURBINE GENERATOR AND AUXILIARIES</u>										
Turbine Generator	NN	NA	NA	QA-2		NN	NA	Non-I	NA	
Main Steam Piping										
o Piping and Valves	NN	B31.1	I	QA-2	6	2/NN	III-1/-	I/-	App. B	
o Safety and Relief Valves	2	III-1	I	QA-2	6	2	III-1	I	App. B	
Main Condenser						NN	NA	Non-I	NA	
Bypass Piping and Valves	NN	B31.1	-	QA-2		NN	B31.1	Non-I	NA	11
Feedwater System										
o Piping and Valves	NN	B31.1	-	QA-2		NN	B31.1	Non-I	NA	9
o Feedwater Heaters	NN	-	-	QA-2		NN	NA	Non-I	NA	
o Feedwater Pumps	NN	-	-	QA-2		NA	NA	Non-I	NA	

NOTES FOR TABLE 3.2-3

1. Safety Classification is that use by the American Nuclear Society, ANS 50 Committee, and relates to other classification schemes as follows:

<u>Safety Class</u>	<u>NRC Quality Group</u>	<u>ASME III Class</u>
1	A	1
2	B	2
3	C	3
NN	D or Augmented D	N/A

2. ASME Code refers to Section and Division of ASME Boiler & Pressure Code as follows:
 - III-1 ASME B&PV Code, Section III, Division 1
 - III-2 ASME B&PV Code, Section III, Division 2
 - VIII ASME B&PV Code, Section VIII, Division 1 or 2, as appropriate
 - B31.1 ANSI B31.1 Power Piping Code
3. Refers to Seismic Category as defined in USNRC Regulatory Guide 1.29.
4. Refers to QA Programs for HWR plants and as contained in Canadian Standards, as follows:

<u>HWR Class</u>	<u>CSA Standard</u>
QA-1 Quality Assurance Program Requirement	Z299.1-1975 Quality Assurance Program Requirements
QA-2 Inspection System Requirements	Z299.2-1975 Quality Control Program Requirements
QA-3 Basic Inspection Requirements	Z299.4-1975 Inspection Program Requirements
	Z299.3-1975 Quality Verification Program Requirements

5. Recommended USA classification is Title 10, Chapter I, Code of Federal Regulations, Part 50, Appendix B - Quality Assurance Program requirements. This is implemented through the ANSI N45.2 series of Standards, and through criteria of ASME B&PV Code, Section III, Divisions 1 and 2.
6. Class 2 applicable to portions of steam and boiler feedwater piping inside containment.
7. Safety Class 3 applies to the spent resin portion only, otherwise NN applies.
8. The Calandria Vault would have to be Safety Class 1 and would be constructed in accordance with the ASME B&PV Code, Section III, Division 2, for Concrete Reactor Vessels and Containments. This classification would be based on its function; that is, the structure containing and supporting the nuclear reactor proper.
9. Feedwater piping is classed as NN beyond containment shutoff valve.
10. Only portions of system supply cooling water to Safety Class components or systems shall be Class B. Other portions of the system shall be Class NN.
11. Valves from main steam header to bypass piping to be Safety Class 2.
12. The end shields, supports and calandria tube sheets provide the support for the Class 1 pressurized tubes, therefore, these parts should carry the same classification as the component supported. Since the calandria shell provides the support for the reactivity mechanisms and inlet nozzles for liquid inspection, it should be reclassified to Safety Class 2.

TABLES 3.2-4 TO 3.2-7 INCLUSIVE

CODES AND STANDARDS MATRICES

The matrices contained herein are provided as an implementation guide of the major codes and standards used in the design, specification preparation and review processes of nuclear power plants.

The classifications used in these matrices are related to the NRC and industry classification systems by the following table:

Classification used in Matrices	1 ⁽¹⁾	2 ⁽¹⁾	3 ⁽¹⁾	D	N	I ⁽²⁾	MC ⁽¹⁾
NRC Quality Group	A	B	C	D	-	-	-
ANSI, ANS	1	2	3	NNS compon- ents do or may contain radio- active material (R.G 1.26)		2 ⁽³⁾	2
ASME Class	1	2	3	-	-	-	MC
IEEE Class	-	1E		-	-	-	-

- Notes: 1) These classes are also designated as seismic Category I
 2) Seismic Category I
 3) Applies to concrete containments only

TABLE 3.2-5 REGULATORY GUIDES

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NOTE: Regulatory Guides apply to Class D and N components only if they are designed for the SSE by the requirements of Regulatory Guide 1.29

TABLE 3.2-6 USNRC BRANCH TECHNICAL POSITIONS

COMPONENT NAME																				
	S	P	V	V	H	C	C	O	C	A	P	C	O	A	O	T	O	T	O	T
	PY	IS	PT	IE	NM	CS														
CLASSIFICATION	1	2	3	D	N	1	2	3	D	N	1	2	3	D	N	1	2	3	D	N
TECHNICAL POSITION																				
AAB 3-2																				
APCSB 3-1	X	X	X	X																
APCSB 6-1	X	X																		
CSB 6-1																				
CSB 6-2																				
CSB 6-3																				
EICSB-3	X	X	X																	
EICSB-4																				
EICSB-10	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
EICSB-13	X	X																		
EICSB-18																				
EICSB-19	X	X																		
EICSB-25	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
MEB 3-1	X	X																		
MEB 5-1	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
RSB 3-1	X	X																		
RSB 3-2	X	X																		
RSB 6-1	X	X																		

TABLE 3.2-7 CONSENSUS CODES AND STANDARDS (SEE FOOTNOTE)

[illegible]

NOTE: The above Codes and Standards are primary documents used in the construction of the subject components. Other standards are required for specifying the complete criteria, some of which are referenced in the primary documents.

3.3 DESIGN CRITERIA FOR STRUCTURES AND EQUIPMENT

Safety-related structures, systems, components, and other equipment related thereto shall be designed to withstand the combined loadings due to the system process conditions and the following loadings, where applicable:

1. Plant design life 40 years
2. Seismic Category I
 - o Safe shutdown earthquake 0.25 g
 - o Operating basis earthquake 0.125 g
3. Ground settlement (plant life)
 - o Absolute settlement 12 inches
 - o Differential (maximum) 10 inches
4. Snow loads 80 lbs/sq ft (max)
5. Wind loads
 - o Operational 130 mph @ 30 ft above ground level, except at coastal sites where it may increase.
 - o Hurricanes To be determined for coastal sites.
 - o Tornadoes 290 mph rotational plus 70 mph translational. Pressure drop of 3 psi at a rate of 2 psi per second.
6. Missile loads
 - o Tornado generated missiles
 - o Internal generated missiles
 - o Site related missiles
7. Flooding protection
8. Containment to withstand effects of jet stream due to postulated high-energy piping.

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4.0 REACTOR PLANT

4.1 Summary Description

The reactor is a horizontal pressure tube device, fueled with approximately 84,000 kg natural uranium dioxide arranged in 380 fuel assemblies. It is heavy water moderated and cooled. Several key reactor characteristics are summarized in Table 4.1-1.

The reactor is installed in a steel-lined concrete vault which is filled with light water for shielding. The main reactor structure is an austenitic stainless steel calandria which is equipped with 4560 Zircaloy calandria tubes as shown in Figure 4.1-1. Concentric with the calandria tubes are Zircaloy pressure tubes which contain the fuel bundles. The pressure tubes pass through end shield regions which are light water cooled steel ball beds.

The reactor is equipped with two shutdown systems: one system consists of 28 vertical cadmium poison rods; the second is a gadolinium nitrate solution in D_2O which can be injected into the moderator through six horizontal perforated tubes propelled by compressed helium. Refer to Section 4.6 for a detailed discussion.

Neutron flux level trim is provided by four vertical solid absorber rods located near the core periphery. Flux shaping is provided by 21 vertical adjuster rods which are designed as varying thickness stainless steel tubes, and also provided by six vertical liquid zone control assemblies which use light water as the neutron poison.

TABLE 4.1-1

Key Reactor Characteristics

<u>Item</u>	<u>Data</u>
Type	Horizontal Pressure Tube
Coolant	Pressurized D ₂ O
Moderator	D ₂ O
Fuel	Natural UO ₂
Number of Channels	380
Reactivity Control:	
Main Method	On Power Refueling and Moderator Poison Control
Trim	H ₂ O Zone Control Assemblies and Vertical Control Absorber Rods
Poison Override	Vertical Adjuster Rods
Shutdown:	
System No. 1	Gravity Shutoff Rods
System No. 2	Moderator Poison Injection
Flux Flattening:	
Axial	Vertical Adjuster Rods
Radial	Differential Fueling and Vertical Adjuster Rods
Flux Control	H ₂ O Zone Control Assemblies

1	CALANDRIA	14	FLUX MONITOR AND POISON INJECTION
2	CALANDRIA SHELL	15	ION CHAMBER
3	CALANDRIA TUBES	16	EARTHQUAKE RESTRAINT
4	EMBEDMENT RING	17	VAULT WALL
5	FUELLING TUBESHEET	18	VAULT COOLING PIPES
6	END SHIELD LATTICE TUBES	19	MODERATOR OVERFLOW
7	END SHIELD COOLING PIPES	20	PRESSURE RELIEF PIPES
8	INLET-OUTLET STRAINER	21	PRESSURE RELIEF DISC
9	STEEL BALL SHIELDING	22	REACTIVITY CONTROL ROD NOZZLES
10	END FITTINGS	23	VIEWING PORT
11	FEEDER PIPES	24	SHUTOFF ROD
12	MODERATOR OUTLET	25	ADJUSTER ROD
13	MODERATOR INLET	26	CONTROL ABSORBER ROD
		27	ZONE CONTROL ROD
		28	VERTICAL FLUX MONITOR

REACTOR ASSEMBLY

FIGURE 4.1-1

4.1-3

4.2 Fuel System Design

4.2.1 Function

The HWR fuel is natural uranium in order to avoid the economic penalty of fuel enrichment in U-235. The purpose of the fuel is to provide a heat source through the fission process. To enhance reactor availability, the fuel is designed to be replaced without shutting down the reactor.

4.2.2 Description

The fuel consists of natural uranium dioxide, compacted and sintered into cylindrical pellets which are stacked in sealed Zircaloy tubes 1.31 cm in diameter by approximately 49 cm long. 37 fuel elements are welded to end-plates to form a fuel bundle. 12 fuel bundles are stacked into each of the 380 pressure tubes in the reactor. In each bundle the outer ring of elements is 1 cm shorter than the others. The total uranium weight in the core is 84,000 kg. Spacing between adjacent elements and between the bundle and the pressure tube is maintained by interelement spacers and bearing pads respectively, which are welded to the Zircaloy sheaths, as shown in Figure 4.2-1.

The operating conditions have changed as the HWR's evolved. Typical fuel parameters are a maximum outside clad temperature of 326°C (619°F), a maximum linear heating rate of 4.30 kw/meter (1.31 kw/ft), and a maximum surface heat flux of 1288 kw/m² (4561 Btu/sec-ft²).

The clad is designed to slump onto the fuel during operation; this demands close control over gap dimensions to avoid damage to the cladding.

The HWR's are designed for on-line refueling. This permits operation of the core with a relatively low fission produce inventory. Excess reactivity

requirements are therefore limited to xenon override and control. This is consistent with the neutron economy which is carefully addressed throughout the design of the HWR; reduced fueling cost is an added bonus.

Refueling is accomplished with two fueling machines. They are remotely controlled to latch onto the chosen pressure tube, one upstream and the other downstream. A new fuel bundle is inserted by the upstream machine; an irradiated one is accepted by the downstream machine and then placed in the spent fuel receiving bay. Usually eight fresh bundles are added upstream at each refueling. The four bundles from the upstream region end up in the downstream region. Thus, the axially central four bundles reside in the core always just half the time that the other eight bundles do. Details of the refueling operations are discussed in Section 9.1.

Because adjacent fuel channels always have opposite flow directions, and refueling is in the direction of coolant flow, burnup - and therefore flux and power distribution - are axially nearly symmetrical.

4.2.3 Design Bases

For the fuel and fuel assembly design data, refer to Table 4.2-1.

4.2.4 Safety Evaluation

4.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the fuel system:

GDC 10 - Reactor Design

4.2.4.2 Assessment

There is insufficient information regarding the fuel system design to properly assess its design with respect to U.S. licensing criteria.

Refueling is not a safety function. The refueling approach appears to be compatible with the General Design Criteria and is therefore licensable. However, there is growing concern over spent fuel accountability, and criteria should be developed to avoid diversion of spent fuel.

4.2.4.3 Recommended Design Change

- o Develop criteria for spent fuel security for on-line refueling.

TABLE 4.2-1Fuel Assembly Design Data

<u>Item</u>	<u>Data</u>
Fuel	
General	
Fuel	Compacted and Sintered Natural UO ₂ Pellets
Form	Fuel Bundle Assembly of 37 Elements
Weight	
Total Weight of UO ₂ in Reactor	2.116 x 10 ⁵ lb
Total Weight of U in Reactor	1.874 x 10 ⁵ lb
Residence Time at 82 percent Capacity Factor, Average	382 days
Pellets	
Quantity	Approx. 29 per element
Form	Cylindrical Pellets with Concave Dished Ends
Diameter	0.4785 in
Stack Length:	
Outer Elements	18.5163 in
Intermediate Elements	18.9100 in
Inner Ring	18.9100 in
Center Element	18.9100 in
Stack Weight:	
Dish Void (Volume %)	1.3%
Density:	
Nominal	661.73 lb/ft ³
Range	652.36 - 671.09 lb/ft ³

TABLE 4.2-1 (cont'd)

Fuel Assembly Design Data

<u>Item</u>	<u>Data</u>
Fuel Element Assembly	
General:	
Form	
Sheath:	
Quantity (per Fuel Bundle)	37
Material	Zircaloy 4
Weight	5.112 lb
Outside Diameter	0.525 in
Thickness	0.016 in
Sheath-Fuel Diametral Clearance	0.0035 in
Sheath Attachment Joints:	
End Caps to Sheath	Resistance Brazed
Bearing Pads to Sheath	Brazed
Spacer Pads to Sheath	Brazed
37-Element Bundle Design	
Channel Specifications:	
Quantity of Fuel Bundles:	
In Reactor	4560
Per Fuel Channel	12
Weight	
UO ₂ per Bundle	47.068 lb
U per Bundle	40.785 lb

TABLE 4.2-1 (cont'd)

Fuel Assembly Design Data

<u>Item</u>	<u>Data</u>
Nominal Cold Fuel Dimensions	
Length of Bundle	19.488 in
Outside Diameter (Over Bearing Pads)	4.031 in
Weight of Bundle	52.183 lb
Cross-Sectional Area, Including Sheath	7.707 in ²
Cross-Sectional Area, UO2 (Mid-Plane)	6.654 in ²
Area Ratio UO2/Coolant	1.255
Outer Element Ring:	
No. of Elements	18
P.C. Diameter	0.341 in
Intermediate Element Ring:	
No. of Elements	12
P.C. Diameter	0.216 in
Inner Element Ring:	
No. of Elements	6
P.C. Diameter	0.117 in
Center Element	1 only
Element-Element Spacing:	
At Mechanical Spacers Between Mechanical Spacers	0.040 in
Element-to-Pressure Tube Spacing	0.040 in

TABLE 4.2-1 (cont'd)

Fuel Assembly Design Data

<u>Item</u>	<u>Data</u>
Maximum Heat Rates and Power Dist'n:	
Neutron Flux Depression	
Nominal Fuel Rating	1.280 KW/ft
Nominal Element Ratings:	
Outer Element Ring	1.312 KW/ft
Intermediate Element Ring	1.040 KW/ft

END VIEW

- 1 ZIRCALOY BEARING PADS
- 2 ZIRCALOY FUEL SHEATH
- 3 ZIRCALOY END SUPPORT PLATE
- 4 URANIUM DIOXIDE PELLETS
- 5 INTER ELEMENT SPACERS
- 6 PRESSURE TUBE

FIGURE 4.2-1 37 ELEMENT FUEL BUNDLE

4.3 Fuel Channels

4.3.1 Function

The fuel channel assemblies provide housing for the fuel assemblies and are an integral part of the high pressure primary heat transport system. Each fuel channel has two end-fittings which allow on-line refueling operations.

4.3.2 Description

A fuel channel assembly consists of a zirconium alloy pressure tube expanded at each end into the hub of an alloy steel end fitting as illustrated in Figure 4.3-1. Each assembly with its contained fuel and heavy water coolant is supported by the end shield lattice tubes through sliding bearings, and partly by the calandria tube/pressure tube annular spacers. The end fittings are designed to allow relative axial movement between the fuel channel assemblies and the lattice tubes due to the thermal expansion and pressure tube creep.

The channel closure consists of a flexible seal disc mounted on a body which locks firmly into the end fitting. A shield plug is locked into each fitting. Both the channel closure and the shield plug can be removed and reinserted by the fueling machine during refueling.

The pressure tubes are classified as ASME Boiler and Pressure Vessel Code, Section III, Class 1 vessels. The zirconium-niobium alloy used is not listed as a Code material but the design stresses are based on the factors of safety specified by the Code. The wall thickness calculations use the methods specified by the Code, and a corrosion and wear allowance of 0.20 mm is provided.

4.3.3 Design Bases

Refer to Table 4.3-1.

4.3.4 Safety Assessment

4.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the fuel system:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 10 - Reactor design.
- GDC 14 - Reactor coolant pressure boundary.
- GDC 15 - Reactor coolant system design.
- GDC 30 - Quality of reactor coolant pressure boundary.
- GDC 31 - Fracture prevention of reactor coolant pressure boundary.
- GDC 32 - Inspection of reactor coolant pressure boundary.

ASME Section III rules for construction of nuclear power plant components, Division 1.

4.3.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with in the design of the fuel channel assemblies.

The fuel channel assemblies are supported at each end by end shield lattice tubes through sliding hearings. Some load transfer between the pressure tubes and calandria tubes is also permitted by toroidal springs which

separate fuel channels from the calandria tubes. In view of the length of the fuel channels (19.5 ft between the calandria tubesheets) it would appear that a seismic event could cause considerable radial deflections of the pressure tube/calandria tube assemblies and possible interference with the free movement of the reactivity control assemblies. However, insufficient information exists to assess the seismic response of the fuel channel assemblies and whether additional support structure are necessary.

The practice of mechanically expanding the pressure tubes into the tubesheet is prohibited by paragraph NB 3671.2 of ASME B&PV Code, Section III, Division 1, "Expanded Joints". Further, it is questionable whether or not the mechanical compression joint connecting the feeder lines to the nozzle assemblies meets the requirements of ASME Section III, Division 1, paragraph NB 3671.7, "Sleeve Coupled and Other Patented Joints".

The zirconium-niobium alloy used for the pressure tubes, while not included in Appendix I to Section III of the ASME B&PV Code, Section III, should not constitute a problem of licensability, providing that information regarding its mechanical properties and adequate justification for its use are supplied.

As shown in Table 4.3-1, the fuel channel design pressure is less than the pressure at the fuel inlet, indicating a negative design margin. Refer to Chapter 5 for further discussion, regarding the primary piping systems.

4.3.4.3 Recommended Design Change

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.

GDC 2 - Evaluate the seismic response of the calandria/pressure tube design.

GDC 14 - Increase design pressure on fuel channel assemblies.

ASME Section III, Design pressure tubes and calandria tubes to comply with Division 1
paragraph NB 3671.2.

TABLE 4.3-1Fuel Channel Design Data

<u>Item</u>	<u>Value</u>
<u>Fuel Channel Assemblies</u>	
Quantity	380
Length:	
Overall including end fittings	35 ft. 6 in.
Cross-Sectional Areas (Cold and Unpressurized):	
Pressure Tube (On Min. I.D.)	13.01 in ²
Fuel Bundle	7.71 in ²
Coolant Flow Area	5.30 in ²
Flow:	
Total Flow (All Channels)	60.3 x 10 ⁶ lb/hr
Maximum Channel Flow	190,000 lb/hr
Coolant Volume (Each Assy.)	2.5 ft ³
Coolant Exit Velocity (Central Channel)	30 ft/s
Temperature:	
Reactor Inlet Header	511.8°F
Fuel Inlet	511.8°F
Fuel Outlet	594°F
Reactor Outlet Header	590°F
Pressures:	
Reactor Inlet Header	1630 psia
Fuel Inlet	1602 psia
Fuel Outlet	1493 psia
Reactor Outlet Header	1449 psia

TABLE 4.3-1 (cont'd)

Fuel Channel Design Data

<u>Item</u>	<u>Value</u>
<u>Pressure Tubes and Spacers</u>	
General:	
Quantity	380
Material	Zirconium - 2.5% Niobium Alloy
Dimensions: (Cold)	
Length Trimmed for Installation (Approx)	20 ft 8 in
Inside Diameter, Minimum	4.070 in
Cross-Sectional Area (Internal)	13.01 in ²
Wall Thickness, Minimum	0.165 in
Average	0.171 in
Weight, Approx.	135 lb
Design Temperature	602.6 F
Design Pressure	1592 psia
Design Stress	22660 psia
Location of Critical Point	72% From Inlet
Pressure Tube to Calandria Tube Spacers	
Quantity (Per Channel)	4
Material	Zr-2.5 Nb-0.5 Cu Alloy

TABLE 4.3-1 (cont'd)

Fuel Channel Design Data

<u>Item</u>	<u>Value</u>
<u>End Fittings</u>	
Quantity	760
Material:	
Body	Type 403 Stainless Steel
Liner Tube	Type 410 Stainless Steel
Bellows Attachment Ring	Carbon Steel, ASTM A148
Closure Seal Insert Ring	Custom 455 Hardened Stainless Steel
Joints:	
Liner Tube to Body	Rolled Joint
Bellows Attachment Ring to Body	Shrink Fit
Closure Seal Insert to Body	Shrink Fit
Feeder Attachment Provision	Grayloc Seal-Bolted Flange
Pressure Tube Attachment	Rolled Joint
Body:	
Dimensions:	
Inside Diameter:	
Minimum	4.442 in
Maximum	5.585 in
Outside Diameter:	
Minimum	6.062 in
Maximum, near Bellows at F/M End	7.000 in 6.705 in
Length	93 in
Weight	300 lb

TABLE 4.3-1 (cont'd)

Fuel Channel Design Data

<u>Item</u>	<u>Value</u>
Design Pressure	1720 psig
Design Temperature	605°F

Shield Plugs

General:

Quantity 760

Material:

Body Type 410 Stainless Steel

Latch 17 - 4PH Stainless Steel

Dimensions:

Diameter 4.03 in

Length of Assembly 38.61 in

Weight (Estimated) 75 lb

Channel Closure

General:

Quantity 760

Material

Body 17 - 4PH Stainless Steel

Seal Disc 17 - 4PH Stainless Steel

Dimensions:

Overall Length 9 in

TABLE 4.3-1 (cont'd)

Fuel Channel Design Data

<u>Item</u>	<u>Value</u>
Outside Diameter of Body	4.75 in
Outside Diameter of Seal Disc	4.65 in
Design Pressure	1720 psig
Design Temperature	605°F
Estimated Weight	28 lb
<u>End Fitting Bearings</u>	
General:	
Quantity per Fuel Channel	
Free End Fitting	2 sets
Fixed End Fitting	2 sets
Material:	
Journals	Tool Steel, AISI Type D2
Sleeves	Tool Steel, AISI Type A2
Operating Temperature:	
Inner	560°F
Outer	210°F
Maximum Bearing Loads:	
Normal Operation:	
Inboard	900 lb
Outboard	1850 lb

TABLE 4.3-1 (cont'd)

Fuel Channel Design Data

<u>Item</u>	<u>Value</u>
Upset Conditions:	
Inboard	4000 lb
Outboard	6200 lb
Operating Clearance:	
Min. Radial Operating Clearance Between Journals and Sleeves	0.015 in
<u>Channel Annulus Bellows</u>	
Quantity	760
Material:	
Bellow	Nickel-Chromium Alloy
Flange	Carbon Steel
Ferrule	Stainless Steel Type 304L
Bellows Assembly:	
Outside Diameter, Maximum	9.20 in
Internal Atmosphere	CO ₂
External Atmosphere	Hot Air
Operating Temperature	590°F
Operating Pressure Across Bellow Wall, Maximum	14 psig
Service Life	30 Years
Axial Extension, Shutdown To Normal Operating	0.62 in
Over Plant Life, Max.	3.62 in

Figure 4.3-1 CHANNEL ASSEMBLY MOUNTED IN END SHIELDS

4.4 Moderator Liquid Poison System

4.4.1 Function

The liquid poison system is used:

- a) To add negative reactivity to the moderator to allow for excess reactivity in new fuel.
- b) To compensate for the absence of fission product xenon during a startup after a long shutdown.
- c) To provide a means of adjusting reactivity in conjunction with other reactivity control devices.

4.4.2 Description

A flow diagram of the Moderator Liquid Poison System is shown in Figure 4.4-1.

The liquid poison system is used to control the reactivity of the heavy water moderator by adding soluble chemicals with large neutron capture cross-sections. Boron, as HBO_3 , and gadolinium, as $\text{Gd}(\text{NO}_3)_3 \cdot 6\text{H}_2\text{O}$, are the chemicals used.

Boron is used to compensate for the lack of neutron absorbing fission products in the fuel at initial plant start-up, as a poison shim during refueling, for long term shutdowns and for any other situation which requires poison on a long term basis.

Gadolinium is used on reactor startup following a long shutdown since it is burned to a lower neutron capture cross section at approximately the same rate as the fission product xenon is produced.

Heavy water is supplied to the Poison Storage tanks from the outlet of the moderator heat exchangers through a delay tank which allows nitrogen-16 and oxygen-19 to decay. Both poison storage tanks have agitators to aid in the preparation of the poison solutions. Separate lines from the mixing tanks carry the poison to the suction of the moderator pumps. Needle valves are manually adjusted to obtain the required rate of poison flow from the tanks and operation of motorized valves start or stop the poison flows as required by operation. Poison addition valves are controlled from the control room.

The system will be constructed to Class 3 Section III of the ASME Code. All components in contact with heavy water are austenitic stainless steel.

4.4.3 Design Bases

Refer to Table 4.4-1.

4.4.4 Safety Evaluation

4.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the Moderator Liquid Poison System:

GDC 26 - Reactivity control system redundancy and capability.

GDC 27 - Combined reactivity control systems capability.

GDC 28 - Reactivity limits.

4.4.4.2 Assessment

The Moderator Liquid Poison System provides an additional means of inserting negative reactivity. It is not relied upon as a safety injection system but must be classified as a system important to safety. An underlying fact concerning this system is that one of its functions is to compensate for the

absence of xenon during a startup after a long shutdown. With such a purpose, it becomes apparent that xenon, a neutron poison, is used as a planned poison during a startup operation following a brief shutdown. Such an operation is highly time dependent and does not allow tight control over a startup operation.

4.4.4.3 Recommended Design Change

GDC 28 - Delete the Moderator Liquid Poison System function to compensate for the absence of xenon during a startup after a long shutdown.

TABLE 4.4-1

Moderator Liquid Poison System

Design Parameters

<u>Item</u>	<u>Value</u>
General	
Long Term Reactivity Control:	
Poison	Boric Anhydride
Solubility Limit in D ₂ O at 20°C	24,000 ppm
Concentration of Boron in the Moderator to Simulate - 28.5 mk	3 ppm
Concentration of Poison in the Poison Tank	7,600 ppm
Rate of Poison Solution Addition to Achieve - 0.5 mk/min	1.5 gpm
Startup Reactivity Control:	
Poison	Gadolinium Nitrate Hexahydrate
Solubility Limit in D ₂ O at 20°C	2,100,000 ppm
Concentration of Gadolinium in the Moderator to Simulate - 28.5 mk*	0.75 ppm
Concentration of Poison in the Poison Tank	3,000 ppm
Rate of Poison Solution Addition to Achieve - 0.5 mk/min	0.9 gpm
Pumps	
Name	Poison Solution Sampling Pumps
Quantity	2
Type	Canned Centrifugal

*-28.5 mk is the reactivity required to compensate for complete absence of equilibrium xenon in the reactor fuel.

TABLE 4.4-1 (cont'd)

Moderator Liquid Poison System

Design Parameters

<u>Item</u>	<u>Value</u>
Duty	Intermittant
Capacity	100%
Pump Data:	
Fluid	Poison Solution in D ₂ O
Flow Rate	1.25 gpm
Operating Temperature	Ambient
Head	15 ft
Design Pressure	100 psig
Design Temperature	200°F
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Power Supply	Class IV
Construction:	
Materials in Contact with D ₂ O	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Tanks	
Name	Poison Mixing Tanks
Quantity	2
Type	Vertical Cylinder with Dished Heads

TABLE 4.4-1 (cont'd)

Moderator Liquid Poison System

Design Parameters

<u>Item</u>	<u>Value</u>
Capacity	14.5 ft ³
Design Conditions:	
Operating Pressure	5 psig
Operating Temperature	100 to 160°F
Design Pressure	100 psig
Design Temperature	200°F
Construction:	
Material:	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Name	Delay Tank
Quantity	1
Type	Horizontal Cylinder with Dished Heads
Capacity	1.2 ft ³
Design Conditions:	
Flow Rate	10 gpm
Operating Pressure	80 psig
Operating Temperature	160°F
Design Pressure	100 psig
Design Temperature	200°F

TABLE 4.4-1 (cont'd)

Moderator Liquid Poison SystemDesign Parameters

<u>Item</u>	<u>Value</u>
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Mixer	
Name	Poison Mixer
Quantity	2
Type	Electric motor-driven agitator, mounted in mixing tank
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Power Supply	Class IV
Construction:	
Material in Contact with D ₂ O	Austenitic Stainless Steel
Applicable Code	None

MODERATOR LIQUID POISON SYSTEM
Figure 4.4-1

4.5 Annulus Gas System

4.5.1 Function

The purpose of the annulus gas is to restrict heat transfer from the fuel channel pressure tubes to the moderator.

4.5.2 Description

A schematic flow diagram of the Annulus Gas System is shown in Figure 4.5-1. The method of distribution to the individual nozzles is shown in Figure 4.5-2.

The fluid employed is dry carbon dioxide gas which is supplied from high pressure storage cylinders. Provisions are made to sample the gas in the system to detect moisture, which would serve as an indication of leakage from the moderator, or more likely, from the heat transport system into the annulus.

A regulated supply of gas is directed to headers and branches at one face of the reactor and is distributed by tubes to the annuli in a series-parallel arrangement. A ring header at the other face of the reactor collects the discharge from the annuli and a branch line directs a flow to the sample station and to the Reactor Building ventilation exhaust.

Normally, flow within the system is restricted to make-up flow. An air to gas heat exchanger cools the discharge before sampling or release to the ventilation exhaust. The system piping is 304 stainless steel and is designed to ASME Section III Class 3 Code requirements.

System pressure is monitored and displayed in the control room and high and low pressures are annunciated. All other measurements and all of the system valves are in normally accessible areas of the plant.

The system is protected against overpressure by rupture discs and safety valves.

4.5.3 Design Bases

Refer to Table 4.5-1.

4.5.4 Safety Evaluation

4.5.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the Annulus Gas System:

GDC 1 - Quality standards and records.

GDC 2 - Design bases for protection against natural phenomena.

GDC 4 - Environmental and missile design bases.

GDC 10 - Reactor design.

GDC 14 - Reactor coolant pressure boundary.

4.5.4.2 Assessment

The Annulus Gas System operates at low pressure and has a high design pressure as indicated in Table 4.5-1. Insufficient information exists to determine the consequences to the moderator in the event of unavailability of the Annulus Gas System.

4.5.4.3 Recommended Design Change

None.

TABLE 4.5-1

Annulus Gas System

Design Parameters

<u>Item</u>	<u>Value</u>
Fluid	CO ₂
Estimated Makeup	1.5 ft ³ /min
Purge Flow Rate	18 ft ³ /min
Operating Pressure	2 psig
Design Pressure	60 psig
Design Temperature	450°F
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

ANNULUS GAS SYSTEM

Figure 4.5-1

4.5-5

ANNULUS GAS SYSTEM

Figure 4.5-2

4.6 Reactivity Control Systems

4.6.1 Shutdown System No. 1 (Shutoff Rods)

4.6.1.1 Function

Shutdown System No. 1 serves as the primary means of terminating reactor operation when certain parameters enter an unacceptable range. Twenty-eight spring-assisted shutoff rods fall into the reactor by gravity inside perforated zirconium alloy guide tubes.

4.6.1.2 Description

The cadmium absorber element is sheathed in stainless steel and supported by a stainless steel cable which is wound on a drum inside the shutoff rod drive mechanism. The drive mechanism is mounted directly above the shutoff rod channel on the concrete shield deck. Refer to Figure 4.6-1 for details.

The cable sheave is driven by an electric motor through a gear train engaged by an electromagnetic clutch. When the clutch is de-energized, the sheave is released and the element falls.

The total negative reactivity introduced by the absorber rods is approximately 80 milli-k. The trip system employs two out of three coincidence logic on the parameters given in Table 4.6-1.

Each trip parameter is measured at three separate locations and each channel is provided with its own Class II power supply. The logic is arranged so that loss of power to a channel results in a channel trip.

A block diagram of Shutdown System 1 is shown in Figure 4.6-2.

4.6.1.3 Design Bases

Refer to Table 4.6-2.

4.6.1.4 Safety Evaluation

4.6.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to Shutdown System No. 1.

GDC 20 - Protection System Functions.

GDC 21 - Protection System Reliability and Testability.

GDC 23 - Protection System Failure Modes.

GDC 25 - Protection System Requirements for Reactivity Control Malfunctions.

GDC 26 - Reactivity Control System Redundancy and Capability.

GDC 27 - Combined Reactivity Control Systems Capability.

GDC 28 - Reactivity Limits.

4.6.1.4.2 Assessment

Shutdown System 1 employs essentially the same design as its counterpart of the light water pressurized water reactor with the exception of the drive system, and as such, there does not appear to be any problem associated with its operation or reliability.

4.6.1.4.3 Recommended Design Change

None.

4.6.2 Shutdown System No. 2 (Liquid Poison Injection)

4.6.2.1 Function

As a backup to Shutdown System No. 1, the Liquid Poison Injection System is provided to rapidly terminate reactor operation by injecting concentrated gadolinium nitrate solution into the bulk moderator through six horizontally distributed nozzles.

4.6.2.2 Description

Shutdown System No. 2, as shown in Figure 4.6-3, consists of one helium storage tank containing approximately 40 ft³ of helium at high pressure connected through air-to-close valves to six poison injection tanks filled with 8000 ppm gadolinium nitrate in D₂O.

This system employs 2 out of 3 coincidence logic on the parameters given in Table 4.6-3. The trip points out for Shutdown System 2 are set higher than those for Shutdown System 1. Refer to Figure 4.6-4 for a block diagram of Shutdown System 2.

The quick opening valves are air-to-close/spring-to-open so that loss of air supply will initiate poison injection. The helium storage tank, quick opening valves and poison tanks are located outside the reactor vault where they are accessible for testing during power operation. Each poison tank can be isolated by manually closing valves in both the gas and poison legs to permit maintenance and testing. Administrative control is utilized to prohibit the closing of all valves and thus losing capability of the system during power operation.

The solution of gadolinium nitrate possesses approximately 600 milli-k of negative reactivity in the moderator when completely dispersed. Operation of 5 out of the 6 tanks is fully capable of terminating reactor operation.

Each poison tank contains a floating solid polyethylene ball. When an injection is initiated, the helium driving gas forces the gadolinium nitrate solution to the calandria. When the tank is empty, the ball is driven to the tank bottom where it blocks off the tank outlet and thereby prevents helium from entering the calandria and over-pressurizing the moderator. The ball is also designed to back seat at the top of the tank to restrict the movement of poison due to the variations in calandria moderator level.

Each tank is connected to a horizontal nozzle in the moderator by a 2½ inch stainless steel pipe. Each injection nozzle assembly consists of a

Zircaloy-2 extruded section containing several hundred injection holes. As shown in Figure 4.6-3, a stainless steel bellows is used to seal the moderator and to reduce sag and vibration on the nozzle and a second bellows to seal against light water intrusion from the shield tank.

There is a liquid to liquid interface between the poison solution and the moderator. This interface is located below each poison tank in the downstream vertical leg of a U-shaped section of a pipe. The interface moves by diffusion from the higher density gadolinium nitrate solution towards the calandria at approximately 0.4 cm/day. Conductivity probes are provided to detect when the interface is approaching the top of the U section.

Upon alarm from the conductivity probes, the injection lines are back-flushed and the associated poison tanks refilled with gadolinium nitrate solution.

4.6.2.3 Design Bases

Refer to Table 4.6-4.

4.6.2.4 Safety Evaluation

4.6.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to Shutdown System 2:

- GDC 20 - Protection System Functions.
- GDC 21 - Protection System Reliability and Testability.
- GDC 23 - Protection System Failure Modes.
- GDC 25 - Protection System Requirements for Reactivity Control Malfunctions.
- GDC 26 - Reactivity Control System Redundancy and Capability.
- GDC 27 - Combined Reactivity Control Systems Capability.
- GDC 28 - Reactivity Limits.

4.6.2.4.2 Assessment

The design of the Shutdown System 2 poses some reliability questions. For example, qualifying a polyethylene ball to seismicity and NDE requirements associated with safety systems is indeed a challenge, if not impossible. Further, administrative control of the valve position is unacceptable for such a system.

Such problems are easily amenable to engineering solutions, and are not considered to be significant.

4.6.2.4.3 Recommended Design Change

GDC 23 - Automatic interlocks are necessary to prevent the isolation of more than one poison tank at any time during the reactor operation.

GDC 26 - The design of the poison tanks should be changed in order to avoid the use of a floating solid polyethylene ball. Two alternate designs are proposed:

- a. A train of three smaller poison tanks connected in series as shown in Figure 4.6-3a with a combined volume of 2.8 cubic feet (excluding volume in connecting and outlet piping). The design would require that the amount of helium gas sent to any one train of poison tanks and its pressure be predetermined such that the interface between helium and the poison solution will always remain above the lower end of the U section when the poison injection is terminated.

- b. The second alternate design proposed for the poison tank consists of a cylindrical tank of about 2.8 cubic feet which employs a guided piston-helical spring assembly as shown in Figure 4.6-3b. When an injection is initiated, the pressure balance line is shut off and the high pressure helium gas drives the piston downward to transfer the poison solution into the calandria vessel. The piston stroke is limited at the lower end to prevent damage to the helical spring which is being compressed during injection. During normal operation the piston is held at the upper end of the tank by the spring tension. The upper and lower chambers are separated by the piston in its normal position, but can be made to communicate through an outside bypass line which is kept closed normally. This line is opened to allow helium passage only for the purpose of backflushing and refilling the poison tank. The gadolinium nitrate solution is drained/refilled through a 3/8 inch pipe connected at the lower part of the U section, upstream of the isolation valve.

4.6.3 Liquid Zone Control System

4.6.3.1 Function

The Liquid Zone Control System varies the quantity of demineralized ordinary water in 14 separate compartments of the 6 vertical liquid zone control absorber assemblies for the purpose of spacial flux control. The ordinary water level in the 14 compartments can be varied either in unison for bulk reactivity control or independently for zonal reactivity control.

4.6.3.2 Description

a. Liquid Zone Control Assemblies

The six vertically oriented zone control assemblies, located as shown in Figure 4.6-10, are divided into fourteen zone control compartments. Each of the two assemblies located on the vertical mid-plane of the reactor is divided into three compartments, while each of the outer four assemblies has two compartments. Each compartment has four feed tubes of which two are used for the inflow and outflow of water, and two are used for the inflow and outflow of the helium cover gas. Figure 4.6-5 shows a typical in-core assembly.

b. Process Flow

The liquid zone control system is designed so that a safe shutdown earthquake cannot result in a decrease in the quantity of water in the compartments. A flow diagram of this system is shown in Figure 4.6-6. In addition to the six zone control assemblies, the system consists of a subsystem for water and gas circulation. Except for the delay tank, all equipment of this subsystem is located in an accessible area of the reactor building.

A constant outflow of water from each compartment permits the quantity of water in the compartment to be controlled by varying the rate of inflow. The outflow from all compartments is delayed for decay of short lived neutron activation products before entering the accessible area of the reactor building.

Continuous water circulation is normally maintained with one of the three pumps provided. The pumps are centrifugal, driven by electric motors on Class IV power supply. The water is pumped from the delay tank to the 14 compartments through individual control valves and also to the helium gas compressors. There are two gas compressors, one of which operates intermittently with the other on standby. The helium for the liquid zone control system is supplied from the moderator cover gas system bottle storage. Helium feed and bleed valves control the gas pressure in the 14 compartments so that the feed valves open as the pressure decreases, following a decrease in the water inventory and the bleed valve opens as the pressure increases when the water inventory increases. The bleed valves return the helium to the delay tank.

The catalytic recombination unit connected in series with the bleed valve reduces the concentration of radiolytic hydrogen in helium to acceptable limits. Flame arrestors are provided at the inlet and outlet of the recombination unit.

c. Control and Instrumentation

The water level measurement in each compartment is accomplished by measuring the differential between the pressure in the bubble

tube extending to the bottom of each compartment, and the pressure in the gas balance header. The instrument output is connected to the station computers. A cathode ray tube display is used to show compartment levels. A computer output controls the compartment inlet water control valve. Manual control stations are also provided. The gas balance header pressure is controlled by feeding helium from the storage tank into the header or by bleeding excess helium from the gas balance header into the delay tank. The pressure difference between the delay tank and the gas balance header is used to control the feed and bleed valves.

Annunciation of system malfunctions or process excursions beyond normal operating limits is provided. The control system is designed to prevent a water level decrease in the compartments following a loss of pressure in the H₂O supply header (e.g. loss of Class IV power). The control computer monitors the H₂O supply header and the helium storage tank pressure. If either of these pressures falls below preset limits, an automatic reactor power setback is initiated.

4.6.3.3 Design Bases

Refer to Table 4.6-5.

4.6.3.4 Safety Evaluation

4.6.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the Liquid Zone Control System:

- GDC 20 - Protection System Functions.
- GDC 21 - Protection System Reliability and Testability.
- GDC 23 - Protection System Failure Modes.
- GDC 25 - Protection System Requirements for Reactivity Control Malfunctions.
- GDC 26 - Reactivity Control System Redundancy and Capability.
- GDC 27 - Combined Reactivity Control Systems Capability.
- GDC 28 - Reactivity Limits.

4.6.3.4.2 Assessment

The Liquid Zone Control System, while not a safety related system, possesses suitable redundancy in components and capability in equipment to perform its function.

4.6.3.4.3 Recommended Design Change

None.

4.6.4 Adjuster Units

4.6.4.1 Function

The adjuster rods are provided to shape the neutron flux for optimum reactor power and fuel burnup, and to provide excess reactivity for overriding xenon poison following power reduction. The reactor is designed to develop full power with all adjuster rods inserted.

4.6.4.2 Description

There are twenty-one adjuster rods in the reactor, located as shown in Figure 4.6-10. A typical adjuster unit is shown in Figure 4.6-7. Each adjuster element is a stainless steel tube varying in thickness lengthwise to

provide optimum neutron absorption characteristics. The elements are guided within the calandria by perforated zirconium guide tubes. They are individually raised or lowered slowly by winching drives mounted on top of the deck, powered by electric motors on command from the control computers. Indication of vertical location is provided and vertical overtravel is limited. The casing for the entire drive unit is sealed, since an effective seal in the supporting cable is not practical.

4.6.4.3 Design Bases

Refer to Table 4.6-6.

4.6.4.4 Safety Evaluation

4.6.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the adjuster units:

GDC 20 - Protection System Functions.

GDC 21 - Protection System Reliability and Testability.

GDC 23 - Protection System Failure Modes.

GDC 25 - Protection System Requirements for Reactivity Control
Malfunctions.

GDC 26 - Reactivity Control System Redundancy and Capability.

GDC 27 - Combined Reactivity Control Systems Capability.

GDC 28 - Reactivity Limits.

4.6.4.4.2 Assessment

Except for the drive system which employs a supporting cable, the design of the adjuster units is essentially the same as for pressurized light-water

reactors, and as such, there appear to be no problems associated with their operation and reliability.

4.6.4.4.3 Required Design Change

None.

4.6.5 Solid Control Absorber Units

4.6.5.1 Function

The solid control absorber units adjust the flux level at times when greater reactivity rate or depth is required than that provided by the Liquid Zone Control System.

4.6.5.2 Description

There are four solid control absorbers in the reactor, located as shown in Figure 4.6-10. The design is essentially the same as that of the shutoff absorber rods (Figure 4.6-1). The absorber element is of the stainless steel clad-cadmium tube type.

4.6.5.3 Design Bases

Refer to Table 4.6-7.

4.6.5.4 Safety Evaluation

4.6.5.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the Solid Control Absorber Units:

GDC 20 - Protection System Functions.

GDC 21 - Protection System Reliability and Testability.

GDC 23 - Protection System Failure Modes.

GDC 25 - Protection System Requirements for Reactivity Control Malfunctions.

GDC 26 - Reactivity Control System Redundancy and Capability.

GDC 27 - Combined Reactivity Control Systems Capability.

GDC 28 - Reactivity Limits.

4.6.5.4.2 Assessment

In view of the similarity of design with the pressurized light water reactors, there do not appear to be any problems associated with the operation and reliability of the solid control absorber units.

4.6.5.4.3 Recommended Design Change

None.

4.6.6 Nuclear Instrumentation

4.6.6.1 Function

Nuclear instrumentation is designed to provide neutron flux information over the full range of reactor operation. To provide total monitoring, three ranges of nuclear detectors are furnished; source range, intermediate range and power range. A minimum overlap of one decade is provided between successive ranges of instrumentation.

4.6.6.2 Description

o Source Range

For initial start-up, the instrumentation comprises two complementary neutron detecting systems. One system has its neutron proportional counters located in a dry horizontal in-core flux detector assembly tube, temporarily left open for this purpose. The other system uses counters located out-of-core in a spare hole in the second shutdown

system ion chamber assemblies. Only the out-of-core counters are required for any subsequent low level start-up after an extended shutdown period.

- o Intermediate Range

At low power, uncompensated ion chambers are used for neutron flux measurements for both the regulating and safety systems. Compensated ion chambers are not required in a heavy water reactor system since adequate gamma discrimination can be obtained by employing appropriate materials and gamma shielding in the construction of the ion chamber housings.

Each channel of the regulating and safety systems consists of an ion chamber which produces a current proportional to neutron flux at the chamber and an amplifier unit which amplifies the ion chamber current and produces linear power, low power and log rate signals. The amplifier unit also contains a high voltage power supply to polarize the ion chambers. The amplifiers feed trip panels containing the trip set point logic and trip relays for the protective channels and the computer multiplexer for the regulating channels.

- o Power Range

In addition to the incore flux detectors discussed in Section 4.6.7, out-of-core chambers are also used to monitor core behavior. Six lead shielded housings are mounted on the outside of the calandria shell, each of which contains three cavities for installation of ion chambers, test plugs and start-up instrumentation. A typical ion

chamber housing and three associated penetration tubes are shown in Figure 4.6-8.

4.6.6.3 Design Bases

Refer to Table 4.6-8.

4.6.6.4 Safety Evaluation

4.6.6.4.1 Applicable NRC Criteria

Refer to Chapter 7.

4.6.6.4.2 Assessment

The design and operability of the nuclear instrumentation is such that no problems of licensability are anticipated.

4.6.6.4.3 Recommended Design Change

None.

4.6.7 Incore Monitoring System

4.6.7.1 Function

The incore monitoring system is designed to provide feedback to the computer controllers to assist in maintaining a flat flux as well as signal any upset condition.

4.6.7.2 Description

In the power range, self-powered incore detectors are used for flux measurements because the ion chambers do not provide the necessary spatial flux information and also because ion chambers are subject to signal variations due to variable concentrations of poison in the moderator.

Two types of incore detectors are used in the reactor with emitters made of platinum and vanadium. Platinum detectors have a mixed response, being sensitive to both neutrons and gammas. Approximately 80% of the detector signal is prompt. Vanadium detectors are neutron sensitive only and have a relatively slow response to flux changes. Platinum detectors are used both in the regulating system and in the two shutdown systems while vanadium detectors are used only in the regulating system for flux mapping. A typical flux detector assembly is shown in Figure 4.6-9.

The low level currents from the incore detectors are suitably amplified and fed to the reactor trip panels or to the control computer input signal multiplexer.

Twenty-six vertical and three horizontal incore flux detector assemblies are installed in the reactor as shown in Figure 4.6-10. Platinum detectors located in the vertical assemblies provide signals for the first shutdown system and the reactor regulating system. Detectors in the three horizontal assemblies provide signals for the second shutdown system. A number of vanadium flux detectors distributed over the 26 flux detector assemblies provide information about neutron flux in each of the fourteen reactor control zones.

4.6.7.3 Design Bases

Refer to Table 4.6-3.

4.6.7.4 Safety Evaluation

4.6.7.4.1 Applicable NRC Criteria

Refer to Chapter 7.

4.6.7.4.2 Assessment

Since the incore detector assemblies penetrate the calandria shell and are arranged in the moderator, there is never any requirements for their removal as in the case of LWR's. The sensitivity of the detectors changes over their lifetime because of such factors as detector burnup, control rod positions, fuel burnup and other operational considerations.

Since the incore detectors are not part of the active safety system, there is no safety function to be fulfilled, and therefore only a question of accuracy of measurements can be raised. Similar detectors are used in LWR's with capability to measure relative flux with an accuracy of $\pm 5\%$.

4.6.7.4.3 Required Design Change

None.

TABLE 4.6-1

SHUTDOWN SYSTEM NO. 1 TRIP PARAMETERS

TRIP PARAMETER	DETECTOR TYPE	SETPOINT	INSTRUMENT* DELAY TIME SECONDS	CONDITIONING PARAMETERS
High Neutron Power	Vertical In-core Detectors	115% Full Power	0.15	-
High H.T.** Pressure	Pressure Trans- mitters	10.58 MPa(g) (1549 psia)	0.21	
Low Gross Coolant Flow	P Trans- mitters	(55% Nominal)	0.40	Log Power 1% Full Power
High Rate Neutron Power	East Ion Chambers	10%/Sec	0.25	
Access Control Key Interlock	-	-	-	-
Manual	-	-	-	-
Start-up ⁺ Count Rate				High (Initial Operation Only)

* Time from trip value to trip signal

** H.T. - Heat Transport System

+ For Initial Start-up Only

TABLE 4.6-2

Shutdown System 1 (Shut-off Rods) Design Data

<u>Item</u>	<u>Data</u>
Reactivity Mechanism Deck	
Material	Carbon steel top and bottom plates and webs, filled with ordinary concrete
Dimensions:	
Length	20 ft
Width	15 ft
Depth	4 ft
Structural Depth-Lower	2 ft 9 in
Penetrations:	
Flux Detectors	26
Liquid Zone Controllers	6
Adjuster Units	21
Shut-Off Units	28
Viewing Ports	2
Solid Control Absorbers	4
Reference Points	3
Shipping Weight	45,000 lb
Shut-Off Units	
Absorber Elements:	
General:	
Quantity	28 (two uniformly distributed banks each containing 14 units)

TABLE 4.6-2 (cont'd.)

Shutdown System 1 (Shut-Off Rods) Design Data

<u>Item</u>	<u>Data</u>
Type	Stainless Steel-Clad Cadmium Tube
Dimensions:	
Outside Diameter	4.45 in
Length, Cadmium Tubes:	
24 units	17 ft 8-3/4 in
4 units	14 ft 3-3/4 in
Weight:	
Longer units	100 lb
Shorter units	78 lb
Insertion Sequence	Simultaneous drop, spring augmented acceleration
Withdrawal Sequence	One bank (14 units) at a time
Withdrawal Time (Each Bank), Min.	130 s (Variable)
Travel	21 ft
Reactivity Worth, Approx.	-55 milli-k
Drive Mechanisms:	
Type	Motor-driven winch with cable and rotary hydraulic damper
Coupling, Winch to Motor	Gear train engaged by a dc Electro-Magnetic friction clutch
Power Requirement	230 Vac, 3 Phase, 60 Hz
Electrical Load	Class 3, 1/10 hp (75 W)
Guide Tubes (In-Vault Section):	
Material	Stainless Steel
Outside Diameter	5-1/4 in
Wall Thickness, Approx.	0.148 in

TABLE 4.6-2 (Cont'd.)

Shutdown System 1 (Shut-Off Rods) Design Data

<u>Item</u>	<u>Data</u>
Guide Tubes (In-Core Section):	
Material	Zircaloy
Outside Diameter	5.022 in
Wall Thickness	0.061 in
Axial Tensioning Load Applied	1200 lb
Thimble:	
Material	Stainless Steel
Outside Diameter	6-1/8 in
Wall Thickness, Approx.	3/16 in

TABLE 4.6-3

SECOND SHUTDOWN SYSTEM

TRIP PARAMETER CHARACTERISTICS

TRIP PARAMETERS	DETECTOR TYPE	SETPOINT	INSTR. DELAY TIME SECONDS
Hi Neutron Power	Horizontal In-Core Detector	130%	0.10
Hi Log N Rate	Horizontal Ion Chambers	25%/Sec	0.20
Manual	—	—	—

TABLE 4.6-4

Shutdown System 2 (Liquid Injection) Design Data

<u>Item</u>	<u>Data</u>
Injection Nozzle Assembly	
Quantity	6
Material	Zircaloy 2
Inside Diameter	2.0 in
Length	24 ft 4 in
Bellows Assembly	
Quantity	6
Material	304 L Stainless Steel
Poison Tank	
Quantity, per Unit	6
Description	Cylindrical Shell Capped at Each End by Heads, with Ball Valve
Location	Outside Wall of Reactor Vault
Material:	
Shell and Heads	304 L Austenitic Stainless Steel
Ball Valve	Solid Polyethylene
Dimensions:	
Vessel, inside Diameter	9.3 in
Length	6 ft 9 in
Ball Valve Diameter	9.2 in
Volume	2.8 ft ³
Connections:	

TABLE 4.6-4 (Cont'd.)

Shutdown System 2 (Liquid Injection) Design Data

<u>Item</u>	<u>Data</u>
Inlet	1 in Sch. 40
Outlet	2-1/2 in Sch. 80
Design Pressure	1500 psig
Operating Pressure:	
During Normal Reactor Operation	8 psig
During Poison Injection	1500 psig
After Poison Injection	480 psig
Design Temperature	150°F
Ambient Temperature	40 to 150°F
Contents:	
During Normal Reactor Operation	8000 ppm of Gadolinium in D2O
During Poison Injection	Helium
Applicable Code	ASME Section III, Class 1
Helium Supply Tank:	
Quantity	1
Description	Cylindrical Shell Capped at Each End by Hemispherical Heads
Location	Wall of Reactor Vault
Material:	
Shell and Heads	304 L Austenitic Stainless Steel

TABLE 4.6-4. (Cont'd.)

Shutdown System 2 (Liquid Injection) Design Data

<u>Item</u>	<u>Data</u>
Dimensions:	
Outside Diameter	3 ft 11-1/2 in
Height	5 ft 4 in
Volume	40 ft ³
Connections:	
Inlet	3/4 in Sch. 80
Outlet	3 in Sch. 80
Design Pressure	1500 psig
Design Temperature	150°C
Normal Operating Pressure	1100 to 1200 psig
Ambient Operating Temperature	40 to 150°F
Contents, During Normal Reactor Operation	Pressurized Helium Gas
Applicable Code	ASME Section III, Class 1
Gadolinium Mixing Tank:	
Quantity	1
Description	Cylindrical Shell Capped at Each End by Dished Heads
Location	Reactor Building
Material	304 L Austenitic Stainless Steel
Dimensions:	
Outside Diameter	4 ft 3/4 in
Height	3 ft 1-3/4 in

TABLE 4.6-4 (Cont'd.)

Shutdown System 2 (Liquid Injection) Design Data

<u>Item</u>	<u>Data</u>
Volume	30 Cu ft
Connections:	
Inlet	3/4 in Sch. 40
Outlet	3/4 in Sch. 40
Design Pressure	100 psig
Design Temperature	200°F
Operating Pressure	40 psig
Operating Temperature	40 to 150°F
Contents	8000 ppm of Gadolinium in D ₂ O and Helium
Applicable Code	ASME Section III, Class 3

TABLE 4.6-5

Liquid Zone Control System Design Data

<u>Item</u>	<u>Data</u>
1. <u>Liquid Zone Control Absorber Assemblies</u>	
Quantity	6 (4 two zone, 2 three zone)
Reactivity Worth (Approx)	6 milli-k
Maximum Rate of Change	± 0.11 milli-k/s
Type	Vertical through-tubes, variable quantity water filling by helium pressure regulation
Material:	
Through-tubes	Zircaloy
Thimble Tubes	Stainless Steel
Length:	
Two Zone Units	43 ft
Three Zone Units	46 ft
Outside Diameter:	
Through-tubes	4.60 in
Thimble tubes	6-1/8 in
Wall Thickness:	
Through-tubes	0.050 in
Thimble tubes	3/16 in
Compartments:	
Quantity	14
Inside Diameter	4.50 in
Capacity, Light Water	14 cu ft
Axial Tensile Load Applied	1400 lb
H ₂ O Flow Rates	
Compartment Inflow	0 to 15 gpm
Compartment Outflow	7.5 gpm
Applicable Code	ASME Section III, Class 3
2. <u>Liquid Zone Control Heat Exchanger</u>	
Quantity	1
Type	Tube in Shell
Duty	Continuous
Heat Transferred	1.5 MW(th)

TABLE 4.6-5 (Cont'd.)

Liquid Zone Control System Design Data

<u>Item</u>	<u>Data</u>
Shell Side Data: (Process Side)	
Fluid	H ₂ O
Flow Rate	210 gpm
Inlet Temperature	151°F
Outlet Temperature	100°F
Design Pressure (Gauge)	240 psi
Design Temperature	200°F
Tube Side Data (Cooling Water Side)	
Fluid	H ₂ O
Flow Rate	375 gpm
Inlet Temperature	77°F
Outlet Temperature	105°F
Design Pressure (Gauge)	150 psi
Design Temperature	200°F
Construction	
Tube Material	Incoloy
Tube Outside Diameter	0.625 in
Shell Material	Carbon Steel
Applicable Codes	
Tube Side	ASME Section III, Class 3
Shell Side	ASME Section VIII
3. <u>Liquid Zone Control Pumps</u>	
Quantity	3
Type	Centrifugal
Duty	Continuous
Capacity	100%
Pump Data:	
Fluid	H ₂ O
Flow Rate	275 gpm
Operating Temperature	160°F
Head	350 ft
Design Pressure (Gauge)	240 psi
Design Temperature	200°F

TABLE 4.6-5 (Cont'd.)

Liquid Zone Control System Design Data

<u>Item</u>	<u>Data</u>
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Power Supply	Class IV
Construction:	
Materials in Contact with Fluid	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
4. <u>Liquid Zone Control Gas Compressors</u>	
Quantity	2
Type	Liquid Ring
Duty	Intermittent
Capacity	100%
Compressor Data:	
Fluids	Helium and H ₂ O
Flow Rate (Standard Conditions)	90 ft ³ /min
Discharge Pressure (Gauge)	150 psi
Pressure Differential	145 psi
Design Pressure (Gauge)	240 psi
Design Temperature	200°F
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Power Supply	Class IV
Construction:	
Materials in Contact with Fluids	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class
5. <u>Helium Storage Tank</u>	
Quantity	1
Type	Vertical Cylinder with Dished Heads
Capacity	160 ft ³

TABLE 4.6-5 (Cont'd.)

Liquid Zone Control System Design Data

<u>Item</u>	<u>Data</u>
Design Conditions:	
Operating Pressure (Gauge)	120 to 150 psi
Operating Temperature	Ambient
Design Pressure (Gauge)	200 psi
Design Temperature	200°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
6. <u>Delay Tank</u>	
Quantity	1
Type	Horizontal Cylinder with Dished Heads
Capacity	250 ft ³
Design Conditions:	
Operating Pressure (Gauge)	5 to 35 psi
Operating Temperature	160°F
Design Pressure (Gauge)	150 psi
Design Temperature	200°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
7. <u>Liquid Zone Control Ion Exchange Columns</u>	
Quantity	2
Type	Vertical Cylinder with Dished Top and Conical Bottom
Volume	9 ft ³
Resin Capacity	7 ft ³
Design Conditions:	
Maximum Flow Rate	37.5 gpm
Direction of Flow	Downward
Operating Temperature	100°F
Design Pressure (Gauge)	240 psi
Design Temperature	200°F

TABLE 4.6-5 (Cont'd.)

Liquid Zone Control System Design Data

<u>Item</u>	<u>Data</u>
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
8. <u>Recombination Unit</u>	
Quantity	1
Type	Palladium Catalyst Enclosed in a Vertical Cylinder with Dished Heads
Design Conditions:	
Operating Pressure (Gauge)	75 to 105 psi
Operating Temperature	300°F
Design Pressure (Gauge)	200 psi
Design Temperature	800°F
Construction:	
Material	-
Applicable Code	-

TABLE 4.6-6

Adjuster Units Design Data

<u>Item</u>	<u>Data</u>
1. <u>Absorber Elements</u>	
Material	Stainless Steel
Length:	
Quantity 6	9 ft 4½ in
Quantity 15	11 ft 3 in
Outside Diameter	3 in
Wall Thickness	1/32 in to 1/8 in
Weight	35 lb
Withdrawal + Insertion Sequence	One bank at a time
Withdrawal + Insertion Time,	
Each Bank	65 Seconds
Poison Override Time	30 Minutes
Total Reactivity Worth, Approx.	14 milli-k
Max Rate of Change (per bank)	.08 milli-k/s
2. <u>Drive Mechanisms</u>	
Type	Motor-driven winch with cable, direct coupling
Power Requirement	230 Vac, 3 Phase, 60 Hz
Electrical Load	Class 3, 1/10 HP (75 W)
3. <u>Guide Tubes (In-Core Section)</u>	
Material	Zircaloy 2
Outside Diameter	3.600 in
Wall Thickness	0.048 in
Axial Tensioning Load Applied	1200 lb
4. <u>Guide Tubes (In Vault Section)</u>	
Material	Stainless Steel
Outside Diameter	3.875 in
Wall Thickness	0.148 in
5. <u>Thimble</u>	
Material	Stainless Steel
Outside Diameter	4.750 in
Wall Thickness	.203 in

TABLE 4.6-7

Solid Control Absorber Units Design Data

<u>Item</u>	<u>Data</u>
1. <u>Absorber Elements</u>	
Quantity	4
Type	Stainless Steel-clad Cadmium Tube
Dimensions:	
Outside Diameter	4.45 in
Length, Cadmium Tube	14 ft 3-3/4 in
Weight	78 lb
Insertion Sequence + Method:	
Reactor Stepback	Simultaneous, free drop, gravity acceleration
Reactor Setback	Simultaneous, motor driven
Withdrawal Sequence + Method	Simultaneous, motor driven
Insertion and Withdrawal Time	Variable (130 s minimum)
Full Travel, Max	21 ft
Reactivity Worth, Approx.	10 milli-k
Max Rate of Change (Driving)	± 11 milli-k/s
2. <u>Drive Mechanisms</u>	
Type	Motor driven winch with cable and rotary hydraulic damper
Coupling, Winch to Motor	Gear Train engaged by a d-c electro-magnetic friction clutch
Power Requirement	230 Vac, 3 Phase, 60 Hz
Electrical Load	Class 3, 1/10 HP (75 W)
3. <u>Guide Tubes (In-Vault Section)</u>	
Material	Stainless Steel
Outside Diameter	5 1/4 in
Wall Thickness, Approx	0.148 in
4. <u>Guide Tubes (In-Core Section)</u>	
Material	Zircaloy
Outside Diameter	5.022 in
Wall Thickness, Approx	0.061 in
Axial Tensioning Load Applied	1200 lb
5. <u>Thimble</u>	
Material	Stainless Steel
Outside Diameter	6-1/8 in
Wall Thickness, Approx	3/16 in

TABLE 4.6-8

Ion Chambers Design Data

<u>Item</u>	<u>Data</u>
General:	
Quantity	6
Designation:	
East Side	IC-1, 2 + 3
West Side	IC-4, 5 + 6
Overall Length (Nominal):	
IC-1, 2 + 3	9 ft 6 in
IC-4, 5 + 6	14 ft 6 in
Access Tubes:	
Material	Stainless Steel
No. of Tubes per Unit	3
Inside Diameter	4.291 in
Wall Thickness	0.120 in
Overall Length:	
IC-1, 2 + 3	7 ft
IC-4, 5 + 6	12 ft
Housing:	
Material:	
Shell	Stainless Steel
Shielding	Reactor Grade Lead
Insert Tubes	Aluminum Alloy

TABLE 4.6-8 (Cont'd.)
Ion Chambers Design Data

<u>Item</u>	<u>Data</u>
Shell Outside Diameter	1 ft 9 in
Wall Thickness	3/8 in
Overall Length	2 ft 6 in
Shield Plugs:	
Material	Aluminum Alloy Bar
Diameter	4 in
Overall Length:	
IC-1, 2 + 3	7 ft
IC-4, 5 + 6	12 ft
No. of Plugs per Tube:	
IC-1, 2 + 3	3
IC-4, 5 + 6	4

TABLE 4.6-9In-Core Flux Detector Units Design Data

<u>Item</u>	<u>Data</u>
Vertical Flux Detector Units	
General	
No. of Assemblies	28
Axial Tensile Load Applied	500 lb
Carrier Tube:	
Material	Zircaloy
Outside Diameter	0.379 in
Length (Nominal), Longest Unit	21 ft 2 in
Wall Thickness	0.030 in
Guide Tube (In-Core Section):	
Material	Zircaloy
Outside Diameter	0.820 in
Wall Thickness	0.040 in
Guide Tube (In-Vault Section):	
Material	Stainless Steel
Inside Diameter	1.089 in
Wall Thickness	0.109 in
Thimble	
Material	Stainless Steel
Inside Diameter	1.472 in
Wall Thickness	0.20 in

TABLE 4.6-9In-Core Flux Detector Units Design Data

<u>Item</u>	<u>Data</u>
Horizontal Flux Detector Units	
General:	
No. of Assemblies	3
Overall Length:	
HFD 1 + 3	37 ft 6 in
HFD 2	36 ft 6 in
Carrier Tube:	
Material	Zircaloy
Outside Diameter	0.379 in
Length	
Wall Thickness	0.030 in
Guide Tube (In-Core Section):	
Material	Zircaloy
Outside Diameter	0.820 in
Wall Thickness	0.040 in
Guide Tube (In-Vault Section):	
Material	Stainless Steel
Inside Diameter	1.089 in
Wall Thickness	0.109 in
Thimble:	
Material	Stainless Steel
Inside Diameter	1.472 in
Wall Thickness	0.2 in
Axial Tensile Load Applied	1000 lb

FIGURE 4.6-1

Shut-off and Solid Control Absorber Units

4.6-39

SHUTDOWN SYSTEM NO. 1 - BLOCK DIAGRAM

FIGURE 4.6-2

4.6-9.4

SHUTDOWN SYSTEM 2

FIGURE 4.6-3

MIDDLETOWN PRESSURIZED HEAVY WATER
REACTOR PLANT
Preliminary Safety Information Document

SCHEMATIC OF THE
MODIFIED SHUTDOWN
SYSTEM 2 (VERSION 1)

Fig. 4.6-3a

4.6-43

SHUTDOWN SYSTEM NO. 2 - BLOCK DIAGRAM

FIGURE 4.6-4

FIGURE 4.6-5

Liquid Zone Control Assembly

FIGURE 4.6-6

LIQUID ZONE CONTROL SYSTEM

FIGURE 4.6-7

Adjuster Unit

TYPICAL ION CHAMBER ARRANGEMENT

FIGURE 4.6-8

FIGURE 4.6-9
Flux Detector Assembly

REACTOR CONTROL AND FLUX MONITOR LOCATIONS

FIGURE 4.6-10

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5.0 REACTOR COOLANT AND CONNECTED SYSTEMS

5.1 Primary Heat Transport System

5.1.1 Function

The main heat transport system circulates pressurized heavy water through the reactor fuel channels to remove heat produced by fission of uranium fuel. The heat is carried by the reactor coolant to the steam generators where it is transferred to ordinary water to form steam, which subsequently drives the turbine generator.

5.1.2 Description

The Primary Heat Transport System (PHTS) consists of 380 reactor pressure tubes, four vertical steam generators, four shaft sealed motor driven primary heat transport pumps, four reactor inlet headers, four reactor outlet headers, one electrically heated pressurizer, and all necessary interconnecting piping and valving.

A schematic diagram of the PHTS is shown in Figure 5.1-1. The system, which is located entirely within Containment, is arranged in two heat transport loops, each with two PHTS pumps, two steam generators, two inlet headers, two outlet headers, one set of 190 inlet feeders and an equivalent number of outlet feeders.

The two loops are connected to a common pressurizer and purification circuit, which are isolated when an emergency condition exist and thereby isolated from one another. Two loops are provided in order to reduce the rate of positive coolant void reactivity insertion in the event of a loss of coolant accident. During such an accident, only one half the total coolant volume can be discharged, assuming no other failures. Refer to Figure 5.1-2 for a composite flow diagram.

A. Temperature and Flow

Feeder flows are matched to individual fuel channel powers to give an equal D₂O steam quality of 4.0 percent by weight for each channel at the reactor outlet headers when the reactor is at full power. Pressure drop causes the D₂O steam quality to increase from about 3 percent at the fuel outlet to 4.0 percent at the reactor outlet headers and to about 4.8 percent at the inlet to the steam generator.

The two figure-of-eight loops provide bidirectional flow through the core such that the flow is in opposite directions in adjacent channels.

Each loop removes the heat from half of the 380 channels in the reactor core. Each loop has one inlet and one outlet header at each end of the reactor core. D₂O is fed to each of the 380 fuel channels through individual feeder pipes from the horizontal reactor inlet headers and heavy water is returned from each fuel channel through individual outlet feeder pipes to the horizontal reactor outlet headers. Individual feeder piping sizes depend on the coolant flow through the particular channel. The maximum power channel has a flow of 190,000 lbs/hr.

D₂O enters each reactor outlet header at an average temperature of 590°F and a quality of 4.0 percent steam by weight.

Two 18 in diameter pipes connect each reactor outlet header to one steam generator. As the heavy water passes through the four steam generators, the heat is transferred to the secondary side and the D₂O outlet temperature is 512 F at full power. Refer to Figure 5.1-3.

Each steam generator is connected to the pump suction of one heat transport pump by a 24 inch diameter pipe. Each heat transport pump delivers heavy water to one reactor inlet header through two 16 inch diameter pipes.

B. Steam Generators

Four identical steam generators with integral preheater transfer heat from the D₂O reactor coolant on the primary side to the H₂O feedwater on the secondary side. The steam generators, as shown in Figure 5.1-4, consist of an inverted vertical U-tube bundle installed in a shell. Steam separating equipment is housed in the upper end of the shell.

The primary side consists of the steam generator head or channel cover, the primary side of the tube sheet and the tube bundle. A baffle separates the inlet half of the steam generator head from the outlet half. The U-tubes are welded to the primary side of the clad carbon steel tube sheet and rolled into the tubesheet. The boiler head is carbon steel and is provided with two manways.

The primary side of the steam generators is in accordance with the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code.

The secondary side of the steam generators consists of the shell, the steam separating equipment, the tube bundle shroud, the secondary side of the tube plate, the secondary side of the tube bundle, the preheater baffles and the tube support plates. Carbon steel is the principle material of construction. The shell is provided with a manway and connections for chemical addition and steam generator blowdown.

Incoming feedwater is pumped into the baffles preheater section and flows over the D₂O outlet portion of the U-tube bundle. The feedwater emerges from the preheater section at saturation temperature and mixes with the recirculated saturated water flowing over the other portion of the U-tube bundle.

Saturated water flows to the tube bundle from the steam separating equipment through an annulus between the tube bundle shroud and the inside surface of the shell. The water passes through holes in the bottom of the shroud and flows over the D₂O inlet portion of the U-tube bundle.

The steam-water mixture rising from the upper end of the U-tube bundle passes through the steam separators. The separated water recirculates to the tube bundle. The steam with less than 0.25 percent moisture by weight leaves the boiler through the outlet nozzle. The water level at the separators is controlled by a combination of level measurement, steam flow measurement and feedwater flow measurement.

The secondary side of the steam generators is designed and manufactured to the requirements for Class 2 components in Section III of the ASME Boiler and Pressure Vessel Code.

Orientation of the steam generators with respect to the reactor and pumps is shown in Figure 5.1-5.

C. Primary Heat Transport Pumps

The Primary Heat Transport Pumps are vertical single stage, single suction, double discharge centrifugal pumps. A sectional view of a heat transport pump is shown in Figure 5.1-6.

Each pump is driven by a vertical, totally enclosed, air-water cooled squirrel cage induction motor. The pump is supported by the motor mount which is bolted to the top of the pump casing. A removable shaft coupling connects the motor to the pump. Removal of the coupling allows sufficient space for the pump seals and bearings to be removed without removing the motor.

The motor is supplied with a flywheel so that the total rotational energy of the pump motor unit prolongs pump operation after loss of motor power and the rate of flow approximately matches the power run-down following a reactor trip.

The motor is supplied with two removable radial bearings and a double acting thrust bearing. The motor bearings are lubricated by cooled and recirculated oil. An oil lift system supplies high pressure oil to both sides of the thrust bearing simultaneously during start-up.

The pump casing consists of a vertical bottom suction nozzle, the main bowl and two horizontal discharge nozzles.

The cover, or stuffing box, is mounted on top of the casing and sealed by a double gasket. Within the stuffing box are the heavy water lubricated bearing, restriction bushing, shaft seals, thermal barrier and the shaft. The main impeller is located at the end of the shaft below the bearing.

Shielding is installed between the pump casing and the pump motor to reduce irradiation of personnel engaged on pump and motor maintenance and other maintenance tasks above the pump casing.

The shaft sealing arrangement consists of three mechanical seals and one back-up seal in series. Each mechanical seal is designed to withstand the full differential pressure. All three seals are housed in a removable cartridge located in the upper part of the stuffing box. The back-up seal is located at the top of the stuffing box. In the event of all three mechanical seals failing the backup seal prevents leakage of D₂O to atmosphere while the pump is stopped or is coasting to a stop.

The gland seal circuit supplies cooled and filtered D₂O for lubricating and cooling the mechanical seals. The D₂O is injected below the lowest mechanical seal at a pressure above pump suction pressure and a temperature of 100 F. Part of the gland seal flow passes into the pump casing through a radial restriction bushing while the remainder flows up through the mechanical seals.

D. Headers

There are four reactor outlet headers, two at each end of the reactor. Each outlet header receives flow from 95 outlet feeders and directs the flow to the two inlet pipes of each steam generator.

There are also four inlet headers, two at each end of the reactor. Each reactor inlet header receives flow from the two pump discharge pipes of one pump and distributes the flow to 95 inlet feeders.

E. Feeders

The 380 feeders at each end of the reactor run from the fuel channels vertically up the face of the reactor and then horizontally across and above the fueling machine area to the reactor headers.

Figure 5.1-3 shows the arrangement of the feeders. The feeders range in size from 3½" schedule 80 pipe to 1½" schedule 80 pipe.

The inlet header to outlet header pressure drop is set by the the maximum pressure drop in an inlet feeder - fuel channel - outlet feeder combination. Each of the remaining feeder-fuel channel combinations is designed for the same inlet header to outlet header pressure drop by selecting combinations of pipe sizes for the inlet feeder and the outlet feeder to obtain the required flow resistance. Each flow is set by the fuel channel power so that 4 percent steam quality is obtained at the outlet header.

F. Piping

All piping in the Primary Heat Transport System is seamless carbon steel ASTM A106 Grade B and is designed and fabricated to ASME Section III Class 1 requirements.

G. Instrumentation and Control

Instrumentation connected to this system is used to annunciate operating parameters and for the control of systems such as feed and bleed, pressure control, emergency core cooling and shutdown cooling. Instrumentation is provided to monitor and/or control the following:

- o Operation of pumps
- o Selected channel flow for generating gross low flow trip signal
- o Reactor inlet and outlet header levels during maintenance of major main circuit components

- o Reactor outlet header pressures for Heat Transport System pressure control and protection
- o Reactor inlet and outlet header pressures to generate an emergency core cooling signal

The power supply for the Main Heat Transport System pumps is the station service transformer with backup provided by the offsite-to-onsite system service transformer. The pump motor is designed to withstand the supply interruption caused by auto-transfer between the supply buses.

Each pump has instrumentation such that a pump start is conditional on sufficient oil pressure being present in the jacking system. Each pump is further equipped with transducers to detect vibration and a signal proportional to vibration velocity is transmitted to the plant computer, which annunciates an alarm on high vibration.

There are no automatic trips other than those associated with the normal motor protection logic, and no interlocking logic between pumps.

Gross low flow is detected by monitoring flow through 12 inlet feeders, 3 on each side of circuit 1 and 3 on each of the inlet sides of circuit 2. Shutdown System No. 1 is tripped when certain combinations of feeder elements detect low flow.

Reactor inlet and outlet header levels are monitored during maintenance of the Primary Heat Transport circuit components in order that adequate cover on the reactor inlet feeders is maintained.

Reactor outlet header pressure is used to control the Heat Transport System pressure. Two sets of triplicated pressure measurements are monitored. One set is used for narrow range pressure control during normal operation, and the second is used for wide range pressure control during startup.

The pressure measurements on the reactor headers are used to initiate Shutdown System No. 1, Emergency Core Cooling and also to open the four pressure relief valves on high primary system pressure.

5.1.3 Design Bases

Refer to Table 5.1-1.

5.1.4 Safety Evaluation

5.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Reactor Coolant and Connected Systems:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 14 - Reactor coolant pressure boundary.
- GDC 15 - Reactor coolant system design.
- GDC 30 - Quality of reactor coolant pressure boundary.
- GDC 31 - Fracture prevention of reactor coolant pressure boundary.
- GDC 32 - Inspection of reactor coolant pressure boundary.

ASME B&PV Section III - Nuclear Power Plant Components

5.1.4.2 Assessment

The Quality Assurance Program and record keeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with in the design and construction of the Reactor Coolant and Connected Systems. The program must be expended to provide assurance that the designed structure, system, and component will function as designed. This requires that the program includes all phases of construction, including the material manufacturer and supplier.

Based on the information available to UE&C at the time of preparing this report it does not appear that the system and component design complies with GDC 2. The method of supporting and restraining the steam generators against the effects of earthquakes does not appear to be applicable for these critical components and the magnitude of a safe shutdown earthquake as assigned in U.S. licensing.

The reactor coolant system piping has not been provided with seismic restraints to assure that the system will remain intact during and following earthquake phenomena.

Provisions have not been incorporated in the design of the piping systems to protect systems from the effect of missiles generated within the containment. In addition, there are no provisions to protect the piping of one reactor coolant loop from the effects of a whipping pipe due to a pipe break in another coolant loop. This lack of providing adequate missile protection through plant design is in violation of GDC 4.

The reactor coolant pump motor flywheel will require an investigation to assure that it will not burst during overspeed caused by a reactor coolant pipe break. A burst flywheel would be a source of high-energy missile inside the containment. USNRC Regulatory Guide 1.14 provides criteria for selection of materials, design, fabrication, testing, inspection, and inservice inspection. This Regulatory Guide is an amplification of the requirements of GDC 4. This problem is not addressed in the PHWR documentation used as a basis for this report.

GDC 14 requirements of having an extremely low probability of reactor coolant leakage is questionable due to the mechanical joining of the feeder pipes to the pressure tubes. The use of expanding the pressure tubes into the inlet and outlet end fittings is questionable as to meeting the "extremely low probability" leak-tightness requirement. Material failure portions of this GDC are addressed under GDC 31.

The reactor coolant system design appears to meet GDC 15 except, as noted in subsection 4.3.4.2, the zirconium fuel channel design pressure is less than the coolant pressure at the inlet header indicating that there is an overstress condition. There is also a question as to the safety implication and potential damage that may occur to the reactor coolant pump motors should an automatic transfer of power sources occur during plant operation.

The use of carbon steel to ASTM A-106, Grade B specification for the reactor coolant piping does not meet the intent of GDC 30 for "highest quality standards" or of GDC 31 for "fracture prevention of reactor coolant pressure boundary". Paragraph NB-2332 of ASME III requires that the lowest service

temperature for piping material shall not be lower than $RT_{NDT} + 100$ F, with certain other provisions. This material could be used but only under strict startup and operating conditions. These startup and operating conditions would require that the pressure-temperature conditions be controlled to assure an adequate margin against brittle failure or crack propagation.

GDC 32 on periodic inspection is not met in the PHWR plant used as the bases for this report. All welds, pressure tubes, fittings, steam generator tubes, pumps, and valves of the reactor coolant pressure boundary require volumetric inspection in accordance with 10 CFR 50.55a, ASME Section XI, numerous NRC Regulatory Guides, and GDC 32. The material surveillance program of this criterion is accommodated in the present PHWR.

The present design does not meet the ASME Boiler and Pressure Vessel Code, Section III, Division 1, in the following areas:

- o Expanded tubes, such as calandria tubes in the tubesheet and the pressure tubes into the end fittings are prohibited.
- o Weld joint design for attaching calandria shell to tubesheet is not acceptable; all weld joints must be fully inspectable either by ultrasonics or radiography. This will involve redesign of the calandria tubesheet and calandria weld joints.
- o Since the support of an ASME component is classified to the same class as the component; and since the pressure tubes are ASME Class 1, the calandria tubes, tubesheets, and concrete vault must all be classified as Class 1 since they support and maintain the geometry of the pressure tubes and fuel elements.

5.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC Criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Redesign supports for all components and piping of the reactor coolant system to provide protection to the components against the maximum earthquake forces.
- GDC 4 - The reactor coolant piping systems and applicable components therein should be provided with restraints to prevent a whipping-pipe from damaging another pipe or essential component. In addition, the reactor coolant pump motor flywheel should be investigated and tested to assure that it cannot become a missile.
- ASME B&PV - Redesign reactor complex to meet the requirements of the ASME Boiler and Pressure Vessel Code. The calandria vault should be designed to ASME, Section III, Division 2.
- GDC 14 - The design of the inlet and outlet end fittings, pressurized tubes, and the feeder pipe connections should be investigated and tested to assure that the mechanical type connection designs will provide "leak-tight" operation.
- GDC 15 - The stress level of the zirconium pressure tubes should be reduced below the minimum allowable for the material. A suitable margin should be provided between the maximum actual stress and the minimum allowable stress. In addition, the electrical systems for the reactor coolant pump motors should

- be redesigned to prevent automatic transfer of power sources during plant operation.
- GDC 30 - The material for the reactor coolant piping systems should be
GDC 31 selected to provide an adequate margin between its nil-ductility transition temperature and the minimum temperature at which the piping systems could be partially pressurized.
- GDC 32 - Provide an inservice inspection program that provides for the examination of every weld and mechanical connection in the reactor coolant pressure boundary. Provisions should be made to allow performing a volumetric inspection of the pressurized tubes and fittings, and of the tubes of the steam generators. In addition, the ISI program should provide for functional testing of valves and pumps.

TABLE 5.1-1

REACTOR COOLANT SYSTEM DESIGN DATA

<u>Item</u>	<u>Data</u>
1. Primary Heat Transport Pump	
Quantity	4
Type	Vertical, centrifugal, single suction, double discharge
Duty	Continuous
Capacity	100%
Pump Data:	
Fluid	D ₂ O
Flow Rate	35,000 gpm
Temperature	511 F
Head	750 ft
Design Pressure	1780 psia
Design Temperature	535 F
Motor Data:	
Flywheel size	30,000 lb-ft ²
Type	Vertical, squirrel cage induction motor
Power Supply	Class IV, 6.714 MW (9000 HP)
Operating Load	5.028 MW (6,740 HP)
Construction	
Materials in contact with fluid	Carbon steel
Applicable Code	ASME Section III, Class 1

TABLE 5.1-1 (cont'd)

2. Steam Generators

Quantity	4
Type	Vertical U-tube with integral steam drums
Duty	Continuous
Capacity	25%
Heat transferred	2064 MW(th) (total for 4)
Tube Side Data:	
Fluid	D ₂ O
Flowrate	60.3×10^6 lb/hr (total for 4)
Inlet Temperature	587 F
Inlet quality	4.8%
Outlet temperature	511 F
Design pressure	1550 psig
Design temperature	600 F
Shell Side Data:	
Fluid	H ₂ O
Steam Flow	82.308×10^6 lb/hr (total for 4)
Feedwater flow	7.63×10^6 lb/hr (total for 4)
Blowdown flow (continuous)	8000 lb/hr (total for 4)
Reheater drains flow	688,800 lb/hr (total for 4)
Reheater drains enthalpy	483.68 BTU/lb
Blowdown enthalpy	487.9 BTU/lb
Feedwater inlet temperature	368 F
Pressure at drum nozzle	681 psia
Temperature at drum nozzle	500 F
Quality at drum nozzle	0.25%
Pressure at TSV	660 psia
Design Pressure	735 psig
Design Temperature	510 F

TABLE 5.1-1 (cont'd)

Construction:

Tube material	Incoloy 800
Tube outside diameter	0.625 inch
Tube wall thickness	.046 inch
Shell material	carbon steel

Applicable Codes

Tube side	ASME Section III Class 1
Shell side	ASME Section III Class 2

3. Reactor Headers

Quantity:

Inlet Headers	4
Outlet Headers	4

Material

ASTM A106 Grade B

Dimensions:

Inside Diameter:

Inlet Headers	14.5 in
Outlet Headers	16.0 in

Wall Thickness:

Inlet Headers	2.25 in
Outlet Headers	2.00 in

Pressure:

Design Pressure:

Outlet Headers	1550 psig
Inlet Headers	1870 psig

Temperature:

Design Temperature:

Outlet Headers	600 deg F
	535 deg F

4. Feeders

Material:

Pipe

ASTM A106 GR.B

Dimensions:

Inside Diameter

TABLE 5.1-1 (cont'd)

Size 1	3.364 in
Size 2	2.900 in
Size 3	2.323 in
Size 4	1.939 in
Size 5	1.500 in
Design Pressure	1870 psig
Design Temperature	600 deg F
5. Piping	
General:	
Material:	ASTM A106 Grade B
Dimensions:	
Boiler Inlet:	
Inside Diameter	15.688 in
Outside Diameter	18.000 in
Pump Suction:	
Inside Diameter	17.438 in
Outside Diameter	20.00 in
Pump Discharge:	
Inside Diameter	13.939 in
Outside Diameter	16.000 in
Pressure, Design:	
Boiler Inlet	1550 psig
Pump Suction	1550 psig
Pump Discharge	1870 psig
Temperature, Design:	
Boiler Inlet	600 F
Pump Suction	535 F
Pump Discharge	535 F

5.1-15

MAIN CIRCUIT
Figure 5.1-1

5.1-16

COMPOSITE FLOW SHEET
Figure 5.1-2

HTS TYPICAL FEEDER ARRANGEMENT
Figure 5.1-3

TYPICAL PRIMARY SYSTEM HEAT EXCHANGER

Figure 5.1-4

PRIMARY HEAT TRANSPORT SYSTEM ARRANGEMENT

Figure 5.1-5

HEAT TRANSPORT PUMP (TYPICAL)

Figure 5.1-6

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY (RCPB)

5.2.1 Function

To prevent an uncontrolled release of radioactivity into the containment with subsequent leakage to the environment, the integrity of the reactor coolant pressure boundary be assured. Such assurance depends on the following:

- 1) design and analysis
- 2) appropriate design margins
- 3) selection of proper material to prevent brittle fracture and corrosion
- 4) quality control
- 5) proper operation

5.2.2 Description

The Primary Heat Transport System components will be designed to the requirement of Section III of the ASME Boiler and Pressure Vessel Code for Class 1 components and designed to withstand the design basis earthquake.

In addition, precautions will be taken in the design to ensure that inadvertent operation of equipment will not increase the severity of transients. The effects of corrosion and other chemical effects, erosion, irradiation, vibration, earthquake, fire, deposition and immersion will also be considered.

The effects of critical speeds, shaft stability, and other aspects of the mechanical performance of pumps are considered in the design for operating conditions. The first heat transport pump will undergo a 500 hour endurance test run with 100 starts and stops prior to installation. All other heat transport pumps must meet 100 hour type tests with 20 starts and stops.

5.2.3 Design Bases

The operating conditions considered in the design of the main heat transport systems are given in Table 5.2-1.

5.2.4 Safety Evaluation

5.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Reactor Coolant Pressure Boundary:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 14 - Reactor coolant pressure boundary.
- GDC 15 - Reactor coolant system design.
- GDC 30 - Quality of reactor coolant pressure boundary.
- GDC 31 - Fracture prevention of reactor coolant pressure boundary.
- GDC 32 - Inspection of reactor coolant pressure boundary.

5.2.4.2 Assessment

The assessment of the reactor coolant pressure boundary is included in the assessment in 5.1.4.2 of this report.

5.2.4.3 Recommended Design Change

The recommended design change for the reactor coolant pressure boundary is included in 5.1.4.3 of this report. In addition, the following should be included in:

- GDC 2 - In addition to recommendations given in 5.1.4.3 for GDC-2, Table 5.2-1 "Transient Cycles" should incorporate the appropriate number of cycles for the operational and safe shut-down earthquakes.

TABLE 5.2-1

TRANSIENT CYCLES

<u>Description</u>	<u>Design Cycles</u>
A. Normal Operation (COD I)	
1. Operation at 100% Power	Continuous
2. Heatup from Cold Shutdown to Hot Standby	500
3. Startup from Hot Standby to 100% Power	3,000
4. Shutdown from 100% Power to Hot Standby	100
5. Cooldown from Hot Standby to Cold Shutdown	500
6. Power Maneuvering at 100% Power	10,000
7. Frequency Control Power Transients at 100% Power	1,000
B. Upset Condition (COD II)	
1. Reactor Trip from 100% Power	500
2. Turbine Trip from 100% Power	500
3. PHTS Pump Trip at 100% Power	100
4. Power Stepback from 100% Power	2,000
5. Power Stepback due to Reactor Power	100
6. Loss of Class IV Power	100
7. Loss of Feedwater to a S. G. at 100% Power	100
C. Emergency Condition (COD III)	
1. Rapid Cooldown and Depressurization	15
2. Rapid Pressurization at 100% Power followed by Reactor Trip	15
D. Faulted Condition (COD IV)	
1. Steam Line Break	1
2. Loss of Coolant Accident	1

TABLE 5.2-1 (cont'd)

<u>Description</u>	<u>Design Cycles</u>
E. Test Condition	
1. Hydrostatic Test	20
2. Hot Conditioning	1
3. Leak Test	10

5.3 CALANDRIA

5.3.1 Function

The calandria vessel and its two end shields form an integral multi-compartment structure which contains the fuel channels, calandria tubes, the in-core components of the reactivity control mechanisms and the heavy water moderator.

5.3.2 Description

The calandria shell is a horizontal austenitic stainless steel cylindrical vessel which contains the heavy water moderator/reflector and supports the various incore components of the reactivity mechanism. There are also four 18 in pressure relief nozzles as well as nozzles for moderator cooling. Cool moderator enters the calandria through two 6 in flow connections at each side of the calandria and returns through two 12 in nozzles at the bottom of the calandria.

Figure 5.3-1 shows the orientation of the calandria of the calandria with respect to the Primary Heat Transport System. Figures 5.3-2 and 5.3-3 show sectional views of the calandria and typical penetration locations.

The calandria shell consists of a main shell 24 ft 11 in in diameter by 13 ft $\frac{1}{2}$ in long with a wall thickness of 1-1/8 in and two subshells 22 ft in in diameter by 37 $\frac{1}{2}$ in long with the same wall thickness. The ends of the calandria have reduced diameters which are welded to circular tubesheets. The tubesheets form the inner end of the end shields. There are two annular plates connecting the two shells, approximately 3/4 inches thick, which stiffen the calandria shell and provides a flexible member at each end to allow for differential mechanical and thermal expansion.

The calandria has 115 nozzles for reactivity mechanisms and pipe connections.

Four 18 inch diameter pressure relief pipes are provided at the top of the calandria to restrict pressure increase in the calandria shell in the event of severance of a pressure tube and calandria tube. The relief pipes extend through the light water shield in the calandria vault to rupture disc located in the shielding deck.

A total of 380 Zircaloy-2 calandria tubes span the calandria axially on a 11.25 in square pitch to form a circular lattice array. The calandria tubes pass through the calandria and have expanded ends that are roll expanded with stainless steel inserts into the calandria tubesheets of the end shield forming high integrity leak tight 'sandwich' joints. The tubes are fabricated from zirconium alloy and have dimensions of 5.08 in inside diameter by .055 in minimum wall thickness and 19.5 ft long. They provide an annular gap which is filled with dry inert gas for thermal insulation purposes around the fuel channel assemblies. Refer to Figure 5.3-4 for details.

The calandria tubes act as axial stays between the two end shields. The calandria tubes are separated from the fuel channels by toroidal (garter) springs which also permit some load transfer between the pressure tube and the calandria tube.

The diameter of the calandria tubes is limited by the lattice pitch and by the size of the pressure tube and the gap needed for insulation between the coolant and moderator fluids.

The thickness of the calandria tubes is determined by consideration of collapse under external pressure and stressing from other applied loads during normal operation conditions and liquid poison injection. The calandria tubes are seam welded and annealed.

Each end shield is composed of a 2 inch thick calandria tubesheet, a 3 inch thick fuelling machine tubesheet, 380 lattice tubes, and a shell 1 in thick and 22 ft 2 inches in diameter. The end shields are 3 ft in overall length, as shown in Figure 5.3-5.

The lattice tubes, which are 8 inches in outside diameter and $\frac{1}{4}$ in in wall thickness, and are welded at each end to the tubesheets. All material is 304 L austenitic stainless steel.

The end shield is filled with carbon steel balls varying from $\frac{1}{2}$ to $\frac{3}{8}$ inches in diameter and demineralized light water (60/40 ratio by activation of equipment in the vault and to permit access during reactor shutdown. The water is circulated through the end shields to provide for removal of heat generated within the end shield by conduction through the calandria tubesheet, and convection from the end fittings containing the primary coolant. The end shield cooling system is described in Section 9.2.

As shown in Figure 5.3-6, an integral support structure is provided which consists of a 1 in thick shell and a 0.75 in thick plate. Both are 304 L stainless steel and are welded to the circumference of each end shield and to the carbon steel embedment. The carbon steel embedment is grouted into the vault end walls. The support structure is designed to allow for differential movements between the reactor assembly and the vault which result from thermal and loading effects.

The support rings which are embedded in the calandria vault walls consist of 1.5 in thick carbon steel plates, 1 in thick carbon steel shell and cooling coils. They support the calandria assembly via the support plate and shell and transmit the loads into the concrete vault. To ensure that the concrete temperature is below 150 F, from heat gained by nuclear heating, both water cooling coils and thermal shield slabs are incorporated in the design. Eight inch thick shielding slabs provide protection against radiation streaming through the air gap.

Details of the embedment ring are shown in Figure 5.3-7.

5.3.3 Design Bases

Refer to Table 5.3-1 for design parameters.

5.3.4 Safety Evaluation

5.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Calandria:

- GDC 1 - Quality standards and records.
- GDC 2 - Design basis for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 14 - Reactor coolant pressure boundary.
- GDC 30 - Quality of reactor coolant pressure boundary.
- GDC 31 - Fracture prevention of reactor coolant pressure boundary.
- GDC 32 - Inspection of reactor coolant pressure boundary.

5.3.4.2 Assessment

The Quality Assurance Program and record keeping required by GDC 1 and

Appendix B of 10 CFR 50 are only partially complied with in the design and construction of the Calandria. The program must be expanded to provide assurance that the Calandria will operate as designed. This requires that the program includes all phases of construction including the material manufacturer and supplier.

The Calandria appears to meet the requirements of GDC 2, but would require verification that the seismic loads are included in the analysis.

GDC 4 is satisfied to the extent of missile protection from internal generated missiles. To complete the evaluation it will be necessary to analyze the effect of any missile that could impinge on the fueling tubesheet then damage the Calandria tubesheet.

GDC 14 applies to the PHWR Calandria. The Calandria is an intimate part of the reactor in that it provides support for the pressure tube, contains the moderator and poison, supports the reactivity control elements, and provides a means of removing reactor fuel decay heat. On the basis of its functions, both during normal and emergency operating conditions, the safety classification must be changed from ASME Class 3 to Class 1. The design of the Calandria does not have weld joints that could be radiographically inspected. In addition, the Calandria tube to tubesheet joining would not qualify as presently designed with rolled and expanded tubes in the tubesheet. Section III of the ASME Code requires that the tubes be welded to the tubesheet.

Assessment of the Calandria with respect to GDC 30 and GDC 31 is discussed above in applicability of GDC 14.

Periodic inservice inspection of the Calandria is required based on its importance to operation and to safety. This will require the periodic examination of all welds and of the Calandria tubes to satisfy the requirements of GDC 32, 10 CFR 50.55a and ASME XI.

5.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - No change recommended pending review of design analysis.
- GDC 4 - No change recommended pending review of missile generation, impact, and consequence analysis.
- GDC 14 - Upgrade Safety Class of Calandria from Class 3 to Class 1. Redesign Calandria to fully comply with ASME B&PV Code, Section III, Division 1, for Class 1 components. Redesign method of joining Calandria tubes to tubesheet.
- GDC 30 - See GDC 14.
- GDC 31 - See GDC 14.
- GDC 32 - Design of Calandria and tube-to-tubesheet design should accommodate an inservice inspection program.

TABLE 5.3-1

CALANDRIA AND END SHIELD DESIGN PARAMETERS

<u>Item</u>	<u>Data</u>
General:	
Overall length	25 ft 6 in
Inside (Core) Length	19 ft 6 in
Dry Weight, Less Shielding Balls	240 ton
With Shielding Balls	430 ton
Calandria - Moderator Net Volume	7760 ft ³
Calandria Tube Lattice	11.25 in
Overpressure Relief Openings at top of calandria:	
Quantity	4
Size	18 in
<u>CALANDRIA SHELL</u>	
Form	Horizontal Stepped Cylinder
Material	All Welded Fabrication of 304L Stainless steel to ASTM A240
Dimensions:	
Main Shell:	
Inside Diameter	24 ft 11 in
Thickness	1.125 in
Inside length	13 ft 2 in
Subshell:	
Inside Diameter	22 ft 2 in
Thickness	1.125 in
Inside length	38 in
Dry Weight, Approx.	30 tons
<u>END SHIELDS</u>	
General:	
Quantity	2
Form	Cylindrical vessel with inner and outer tubesheets filled with steel ball shielding and penetrated by 380 lattice tubes.

TABLE 5.3-1 (cont'd)

Material:

Vessel	Type 304L Stainless Steel to ASTM A240
Tube Sheets	Type 304L Stainless Steel to ASTM A240
Filler	Carbon Steel Balls
Lattice Tubes	Type 304L Stainless Steel

Weight, Each (Approx):

Unfilled	71 tons
Filled	166 tons
Overall length (including both tube sheets)	36 in
Inside Diameter	22 ft 2 in

Thickness:

Shell	1 in
Tube Sheet:	
Inner (Calandria Side)	2 in
Outer (Fuelling Machine Side)	3 in
Lattice Arrangement	380 tubes on 11.25 in square pitch
Radius of Outermost Lattice Position	10 ft 3.49 in
Minimum Tubesheet Rim Ligament	5.51 in

Lattice Tube;

Inside Diameter	7.5 in
Outside Diameter	8 in

Shield Coolant:

Type	Demineralized Light Water
Inlet Temp.	140°F
Outlet Temp.	150°F
Design Pressure	90 psig 70 psig due to balls;
Operating Pressure	30 psig remainder water
Design Temperature	150°F
Total Flow through both Shields	2900 gpm

CALANDRIA TUBES

General:

Quantity	380
Material	Zircaloy-2 Annealed

TABLE 5.3-1 (cont'd)

<u>Item</u>	<u>Data</u>
Dimensions:	
Length (overall installed)	19 ft 8 in
Inside Diameter, Minimum	5.077 in
Wall thickness, minimum	0.054 in
Outside Diameter, Maximum	5.209 in
Average Weight Each	49.8 lb
<u>END SHIELD SUPPORT</u>	
General:	
Quantity	2
Form	Cylindrical ribbed shell grouted in concrete vault end wall with circumferential shell and support plate connected to end shield outer tube sheet
Material:	
Embedded parts	ASTM A36 Carbon Steel
Circumferential shell	Weldment of A36 steel and 304L Stainless Steel to ASTM A240
Diaphragm Plate	Type 304L Stainless Steel to ASTM A240
Dimensions:	
Diameter Overall	27 ft 9 in
I.D. Circumferential Shell	22 ft 6 in
Thickness of Support Plate	0.75 in
Weight, Each	30 tons

- 1 MAIN STEAM SUPPLY PIPING
- 2 BOILERS
- 3 MAIN PRIMARY SYSTEM PUMPS
- 4 FEEDERS
- 5 CALANDRIA ASSEMBLY
- 6 FUEL CHANNEL ASSEMBLY
- 7 FUELLING MACHINE BRIDGE
- 8 MODERATOR CIRCULATION SYSTEM

Figure 5.3-1

CONCRETE CALANDRIA VAULT
Figure 5.3-2

5.3-10

REACTOR GENERAL ASSEMBLY (SECTION)

Figure 5.3-3

SCHEMATIC OF INSTALLED CALANDRIA TUBE
Figure 5.3-4

END SHIELD DIMENSIONS AND DETAILS
Figure 5.3-5

5.3-13

CALANDRIA VESSEL BOUNDARY

Figure 5.3-6

EMBEDMENT RING
Figure 5.3-7

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Main Moderator System

5.4.1.1 Function

The function of the heavy water moderator in the calandria is to slow down the high energy neutrons liberated in the fission process to the thermal neutron range. Other functions of the moderator are to provide:

- a) cooling
- b) purification and control of soluble neutron poisons
- c) back-up heat sink in event of loss of ECCS

5.4.1.2 Description

A flow diagram of the main moderator system is shown in Figure 5.4-1. The system is comprised of two 100% capacity pumps and two 50% capacity heat exchangers. The pumps are arranged in parallel and are in series with the two heat exchangers, which are in parallel with respect to each other. Connections are provided for purification, liquid poison addition, heavy water collection, D₂O supply and sampling systems.

The main moderator system pumps, valves and heat exchangers are in a compact arrangement at approximately grade elevation to one side of the calandria vault. The pump suction lines and heat exchanger outlet lines are anchored to a penetration seal where they pass through the calandria vault concrete to reduce the probability of loss of vault shielding water.

The moderator pump motors are connected to the Class IV power supply. In addition, each pump has a pony motor capable of driving the pump at 25% speed and connected to the Class III power supply. In the event of a failure of the Class IV power supply a pony motor will start and operate

a moderator pump at lower speed so that the circulation of heavy water is continued. The cooling water supply to the heat exchangers also continue at a lower flow following a failure of Class IV power. The rate of heat removal is sufficient to limit the increase of moderator temperature in the calandria to an acceptable value during a failure of Class IV power and subsequent reactor shutdown.

The temperature of the D_2O in the calandria is held constant at approximately 150 F by circulating the water from the bottom of the calandria through the main moderator heat exchanger and returning it through nozzle jets inside both sides of the calandria at the horizontal diameter. Cooling water for the heat exchanger is provided by the recirculated cooling water system.

All components are austenitic stainless steel with the exception of the heat exchanger tubes, which are made from titanium.

All pressure retaining components are in accordance with the requirements for Class 3, ASME B&PV Code, Section III.

The only feedback control loop for the main moderator system is used to control the D_2O temperature at the calandria outlet.

A control program in the station digital computers positions the heat exchanger cooling water valves to maintain a constant temperature at the calandria outlet. The feedback signal is the median signal from three temperature sensors located in the calandria outlet piping. Temperature

sensors located at the outlet of the heat exchangers provide control room indication and alarm on high temperature. The alarm setpoint is a function of the reactor power.

Other instrumentation includes:

- a) calandria level
- b) calandria head tank level
- c) pump suction and discharge pressure
- d) pump motor temperature
- e) pump bearing and shaft vibration

The pumps in the main moderator system are electric motor driven, vertical shaft, double suction, centrifugal pumps. Each of the two pumps is rated at 100% of the normal heavy water flowrate. The pumps are designed to function during and following a safe shutdown earthquake, and are controlled from the main control room. Normal operation is one pump running and the other on standby. Failure of the running pump causes the standby pump motor to start automatically.

Each pump main motor is directly coupled to the pump shaft so that the pump and the motor operate at the same speed. The main motors are three phase induction motors connected to the Class IV power supply and are totally enclosed and cooled by a water cooled heat exchanger. Heaters are provided to prevent condensation of moisture inside the enclosure when a main motor is not operating. Each pump pony motor is directly coupled to the main motor shaft and drives the pump at 25% of the main motor speed. The pony motors are three phase induction motors connected to the Class III

power supply and are totally enclosed and fan cooled. Heaters are provided to prevent condensation of moisture inside the enclosure when a pony motor is not operating.

Double mechanical seals operating in series-tandem are provided on each of the pump shafts. One of the two mechanical seals is capable of sealing the pump shaft in the event of failure of the other seal. The mechanical seals are cooled by a heavy water flow from the pump discharge. An atmospheric seal is designed to function as a back-up shaft seal in the event of failure of both of the mechanical seals.

The heat exchangers in the main moderator system are U-tube-in-shell type mounted vertically so that the tube sheet is at the lower end. Each of the two heat exchangers is rated at 50% of the maximum moderator system heat load and is capable of up to 100% of the normal heavy water flow rate. The tube side is designed in accordance with the requirements of Section III of the ASME Code, Class 3, the shell side to Section VIII.

5.4.1.3 Design Bases

Refer to Table 5.4-1.

5.4.1.4 Safety Evaluation

5.4.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Main Moderator System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.

5.4.1.4.2 Assessment

The main moderator system and its associated cooling systems are not classified as safety related under Canadian General Design Criteria. Therefore, applying simultaneous safe-shutdown earthquake with loss-of-coolant, no credit can be taken in the safety evaluation.

The Quality Assurance Program and record keeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with in the design and construction of the Main Moderator System. The program must be expanded to provide assurance that the designed structure, system, and component will function as designed. This requires that the program includes all phases of construction, including the material manufacturer and supplier.

The Main Moderator System appears to meet the requirements of GDC 2. This would be verified through a review of design details of required seismic restraints and that the seismic loads are included in the Analysis.

From the information available, there does not appear to be any provisions to protect the redundant parts of the system from internally generated missiles as required by GDC 4.

5.4.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.

Change the system from Class 3 to ASME Class 2.

GDC 2 - Present design should be reviewed for including seismic loads into the design. Components and piping systems should be reviewed for seismic restraints.

GDC 4 - Provide protection whipping pipes and other missiles for redundant components of the Main Moderator System.

5.4.2 Moderator Purification System

5.4.2.1 Function

The Moderator Purification System performs the following functions:

- a) maintains the purity of the heavy water,
- b) controls the pH
- c) adjust the concentration of soluble poisons in the moderator.

5.4.2.2 Description

A flow diagram of the Moderator Purification System is shown in Figure 5.4-2. Flow through the system varies between 6.6×10^4 lb/hr and 13.2×10^4 lb/hr depending on the number of ion exchange columns being used. The normal operating temperature of the purification system is maintained at approximately 110 F by the Moderator Purification Heat Exchanger.

The system is located in the basement of the Service Building.

The ion exchange columns are designed to hold approximately 7 ft³ of ion exchange resin. The inlet and outlet connections of the column will be provided with 0.008 inch open-slot wedge-wire screens, edge-type screens designed to minimize plugging and to withstand high pressure drops. In the unlikely event of screen blockage, the column design will permit withdrawal of either inlet or outlet screen. The column incorporates a conical bottom to facilitate resin removal by resin slurry. The diameter of the column will be approximately 2 ft and the overall height about 5½ feet.

The purification filter is a cartridge type and is intended to remove any insoluble debris of particle size larger than 5 microns from the moderator system.

The purification system cooler cools the purification flow which originates upstream of the moderator heat exchangers from 160 F to 110 F before the D₂O enters the ion exchange columns.

The material of construction for both the ion exchange columns and the filter, as well as all the valves, strainers and piping is stainless steel, Type 304 or 304L.

The strainer at the outlet of the system is used as a backup and prevents any ion exchange resin that might escape through the resin retaining screens from entering the moderator system. The strainer screen will retain any particles larger than 0.2 mm. The resin bead size is 0.38-0.60 mm.

A conductivity cell, as well as a hypodermic sample pot is located on a sample line downstream from the ion exchange columns to enable the operators to determine when the resin in the column is spent.

The five ion exchange columns will be used as follows: One ion exchange column of mixed bed resin, having equal capacities for anions and cations, and taking a flow of about 125 gpm at 110 F provides adequate cleanup for corrosion products. The resin used would not alter the boron content of the system, but would remove gadolinium if it were present. The clean-up half-life for the above flow will be approximately 6.2 hours.

Two ion exchange columns containing a mixture of cation and anion resin in a ratio 1:6 will be used for boron removal from the moderator system.

A flow control valve located in the purification system return line enables the operators to remove boron from the moderator system at a con-

trolled rate. Poison may be added by valving in a column already saturated with boron which will release boron to the moderator more slowly than is possible with the liquid poison system.

The remaining two ion exchange columns will be used for gadolinium removal (loaded with a 1:1 mixture of cation and anion resin). Gadolinium removal will be required in two cases after normal addition of gadolinium to meet reactivity demands, such as xenon transients and after a poison injection shutdown. In the first case, removal using only one column will be satisfactory with a flow of 125 gpm at 110 F and a half-life of 6.2 hours.

In the second case, a clean-up within the poison-out period is desirable. A clean-up time of 24 hours is considered adequate, and two columns in parallel are used; the resulting flow of 250 gpm at 100 F gives a clean-up half-life of 3.1 hours.

The ion exchange vessels are individually shielded with 2 ft thick ordinary concrete block walls. Above the vessels is the concrete floor which will be 4.5 ft thick. Above each vessel there is a removable slab of the same thickness to allow access to one vessel, if required, without exposing all vessels to the room. A shielding wall 2.5 ft thick separates the equipment from the accessible area of the room. All the valves and strainers are located behind the shielding wall. The purification system cooler is also located behind the shielding wall. The connections on the columns for resin addition and resin removal is in the accessible area.

A monorail is provided in the room directly above the center line of the ion exchange vessels and filter in order to remove the concrete slabs above the vessels and also the vessels themselves using a shielded flask.

5.4.2.3 Design Bases

Refer to Table 5.4-2 for design parameters.

5.4.2.4 Safety Evaluation

5.4.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Moderator Purification System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 50 - Containment design basis.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 53 - Provisions for containment testing and inspection.
- GDC 54 - Piping systems penetrating containment.
- GDC 56 - Primary containment isolation
- GDC 60 - Control of releases of radioactive materials to the environment.

5.4.2.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with in the design and construction of the Moderator Purification System. The program must be expanded to provide assurance that the designed structure, system, and compon-

ent will function as designed. This requires that the program includes all phases of construction, including the material manufacturer and supplier.

Those sections of pipe that pass through the containment and the valves that provide containment isolation should be reclassified from Class 3 to ASME Class 2.

The Moderator Purification System appears to meet the requirements of GDC 2. This would be verified through a review of design details of required seismic restraints and that the seismic loads are included in the analysis. In addition, the removable shields over the equipment in the reactor service building and the monorail and its hoist should be investigated for their seismic capability. Both the shield covers and monorail should meet seismic Category I construction.

With the possible exception of the movable shields and monorail, GDC 4 appears to be complied with for the Moderator Purification System.

Since the Moderator Purification System penetrates containment GDC 50, 51, 53, 54, and 56. Valves must be added to the piping that penetrates containment, one inside and one outside, and capable of automatic operation. In addition, the valves and piping that penetrates containment must meet the fracture toughness requirements of GDC 51.

The system appears to meet the objectives of controlling release of radioactivity to the environment in accordance with GDC 60 requirements. A possible exception to this is the possibility of having the removable shields or monorail dropping onto the ion exchange vessels.

5.4.2.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
The containment isolation valves and the portions of piping that penetrate containment should be ASME Class 2.
- GDC 2 - The component supports, components, and piping design should be verified that they meet the seismic requirements. In addition, the removable shields and monorail and its hoist should be analyzed that they will not become missiles during an earthquake.
- GDC 4 - See GDC 2, above.
- GDC 50 - Piping penetrating containment should be ASME Class 2 and meet fracture toughness requirements of GDC 51. In addition, automatic operating isolation valves should be added to each line penetrating containment, one inside and one outside of containment. The valves should be ASME Class 2 and meet fracture toughness requirements of GDC 51.
- GDC 51 - See GDC 50, above.
- GDC 53 - No change.
- GDC 54 - See GDC 50, above.
- GDC 56 - See GDC 50, above.
- GDC 60 - See GDC 2, above.

5.4.3 Moderator Cover Gas System

5.4.3.1 Function

The moderator cover gas system controls the concentration of deuterium gas by catalytically recombining deuterium and oxygen gases to form heavy water. Helium is used as the cover gas for the moderator system.

5.4.3.2 Description

During normal operation, radiolytic decomposition of the moderator generates deuterium and oxygen gases in the main moderator system. A moderator cover gas system, using helium as the cover gas, guards against explosive concentrations of these gases; see Figure 5.4-3.

The system is connected to the reactor at the highest elevation of the cover gas: at the moderator pressure relief tubes, just inside of the rupture discs. Two 100% capacity diaphragm type compressors in parallel pass the gas through a single heater/dryer to two 100% capacity palladium catalytic recombiners. Flame arrestors are located upstream and downstream of each recombiner. Two pressure relief valves are set to prevent pressure of the lowest row of calandria tubes from exceeding design value in the case of maximum moderator pressure; i.e., after operation of the poison injection shutdown system. Connections to a helium bottle station are provided. The system components will be designed to ASME Section III, Class 3.

5.4.3.3 Design Bases

See Table 5.4-3.

5.4.3.4 Safety Evaluation

5.4.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Moderator Cover Gas System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 20 - Protection system functions.
- GDC 21 - Protection system reliability and testability.
- GDC 22 - Protection system independence.
- GDC 23 - Protection system failure modes.
- GDC 24 - Separation of protection and control systems.
- GDC 25 - Protection system requirements for reactivity control malfunctions.
- GDC 26 - Reactivity control system redundancy and capability.
- GDC 27 - Combined reactivity control systems capability.
- GDC 28 - Reactivity limits.
- GDC 29 - Protection against anticipated operational occurrences.

5.4.3.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied within the design and construction of the Moderator Cover Gas System. The program must be expanded to provide assurance that the designed systems and components will function as designed. This requires that the program includes all phases of construction, including the material manufacturer and supplier.

The Moderator Cover Gas System appears to meet the requirements of GDC 2. This would be verified through a review of design details of required seismic restraints and that the seismic loads are included in the analysis.

An evaluation of the components of the system will be required to assure that adequate redundancy and protection is provided against the effect of internally generated missiles. This is required to meet the requirements of GDC 4.

The cover gas system provides a protective function and a means of removing the formed deuterium gas in the Moderator Calandria. On this basis, GDC 20 through 29 applies to this system. Sudden depressurization of the cover gas could cause boiling of the moderator and in turn effect the reactivity.

Overpressurization of the system could cause excessive external pressure on the Calandria tubes, tending to buckle or crimp the tubes over the pressure tubes.

The system, as now designed, is provided with two overpressure activated valves in series in the helium supply line to prevent a pressure surge. No redundant helium supply is provided to assure a constant supply of cover gas.

Determination of deuterium concentration in the cover gas is accomplished manually on a periodic basis to verify the proper operation of the deuterium-oxygen recombiners.

5.4.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - The component supports, components, and piping design should be verified that they meet the seismic requirements.
- GDC 4 - The layout and design of the system should be verified that adequate redundancy and protection against internally generated missiles is provided.
- GDC 20 o Provide redundant helium supply and control valves to Calandria thru
- GDC 29 head tank and to the Calandria.
 - o Provide a deuterium detector in the moderator cover gas annulus to show continuous deuterium indications in the Control Room and activate alarms at predetermined setpoints.
 - o Provide helium pressure indications in the Control Room and activate alarms at predetermined hi-low set-points.

5.4.4 Moderator D₂O Collection System

5.4.4.1 Function

The purpose of the Moderator D₂O Collection System is to recover the heavy water that may leak from the moderator process auxiliaries.

5.4.4.2 Description

The Moderator D₂O Collection System flow diagram is shown in Figure 5.4-4. The system is located inside the reactor building in an area accessible during normal operation and is designed in accordance with the Class 3 requirements of the ASME Code, Section III. The wetted parts of the system are generally of Type 304 austenitic stainless steel.

The system consists basically of a collection tank, leakage flow indicators, transfer pump and associated piping.

D₂O leakage from the moderator pump seals flows by gravity through the D₂O collection lines to the collection tank. Local visual indication of the source and flow rate is provided.

The transfer pump may be used to pump the contents of the collection tank to the moderator system if the sample shows the water to be of reactor grade, or to drums if the water is downgraded.

Local and control room indication of collection tank level is provided. High level is annunciated. Pressure gauges are provided at pump suction and discharge. Controls for the pump and a motorized discharge valve are provided.

5.4.4.3 Design Bases

See Table 5.4-4.

5.4.4.4 Safety Evaluation

5.4.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Moderator D₂O Collection System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.

5.4.4.4.2 Assessment

Although the Moderator D₂O Collection System has no safety function, it is located inside the containment and must be seismically qualified so that no part of the system can become a missile. Since the system has been classified as ASME Class 3, and with seismic requirements, the system must meet the requirements of GDC 1 for a Quality Assurance Program, GDC 2 for seismic analysis and qualification, and GDC 4 for missile protection.

5.4.4.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - The component supports, components and piping design should be verified that they meet the seismic requirements.

GDC 4 - No changes provided that GDC 2, above, is met and that no component of the system can become a missile that could compromise safety-related systems in the containment.

5.4.5 Moderator Deuteration and De-Deuteration System

5.4.5.1 Function

Ion exchange resins are used in the moderator purification system to remove corrosion products, ionic impurities and liquid poisons (boron and gadolinium) that may be injected into the moderator for reactivity control purposes.

When received from the supplier, the resins contain about 55 percent by weight ordinary water (H_2O). If the resins were used in the as-received condition, the H_2O would downgrade the heavy water. To avoid such downgrading, the H_2O is replaced by D_2O in a deuteration process. Deuterization is the replacement of H_2O in the resin by D_2O by a slow upward flow of D_2O through the resin bed. During the deuteration process, some heavy water is down-graded. This water will be collected and upgraded for further use.

When the resins in the ion exchange columns are spent, they must be replaced and slurried to the resin disposal tanks. Before this is done, all of the heavy water contained in the ion exchange resins must be recovered. This is done by replacing the D_2O with H_2O in a de-deuteration process. De-deuteration is the replacement of D_2O in the resin by H_2O by a downward flow of H_2O through the resin bed. As in the deuteration process, some

heavy water will also be downgraded during the de-deuteration process.

This water will also be collected and upgraded for further use.

5.4.5.2 Description

The Moderator Deuteration and De-deuteration system is shown in Figure 5.4-5.

The deuteration and de-deuteration tanks are provided with removable screens. These screens are capable of withstanding high differential pressures and are designed to minimize plugging. Both the D_2O and H_2O tanks incorporate a conical bottom to facilitate resin removal by resin slurry. The diameter of the tanks will be approximately 2 ft and the overall height about $5\frac{1}{2}$ ft. The tanks are designed to hold approximately 7 ft^3 of ion exchange resin.

Both the D_2O tank and the H_2O tank have a capacity of approximately 15 ft^3 . These tanks are provided with multiple position (4 for the D_2O head tank and 5 for the H_2O head tank) level switches and alarms. Both tanks are located above the deuteration and de-deuteration tanks so that gravity flow can be used for deuteration and de-deuteration. The level switches in the tanks will operate the automatic ON-OFF control valves located in the deuteration tank outlets.

The pump in the system has a capacity of about 25 gpm and is the canned type in order to eliminate seal leakage problems. The pump is used for D_2O service only. The strainers at the pump suction and discharge lines prevents any resin from entering the pump. All the

strainers in the system, as well as the wedge-wire screens, retain any particles larger than 0.2 mm. The resin bead size is 0.38-0.60 mm.

The material of construction for all the tanks and the pump, as well as all the strainers, valves, flexible hoses and piping is Type 304, 304L or 316 stainless steel, and suitable synthetic elastomers.

All the valves in the system except the two located in the main lines to the storage tanks in the D₂O cleanup system are manual valves. These two valves are ON-OFF automatically operated control valves. Diaphragm valves are used for D₂O service and ball valves where resin transfer is involved.

The deuteration and de-deuteration processes can be carried on at the same time, independently of each other.

At the end of the deuteration and de-deuteration operation, the deuteration tank will be full of heavy water and the de-deuteration tank will be full of light water. The heavy water from the deuteration tank will be transferred to the de-deuteration tank using instrument air. The H₂O in the de-deuteration tank is then transferred to the resin disposal vault using instrument air, before D₂O is transferred to it.

Generally, flexible hose connections are provided at every point where it is possible to mix D₂O with H₂O in order to eliminate inadvertent downgrading of D₂O.

The moderator deuteration and de-deuteration system is designed to conform with the ASME Boiler and Pressure Vessel Code, Section III.

Only the de-deuteration tank is shielded with light concrete 4.6 ft thick.

One side of the shielding is removable to allow access to the vessel.

The resin slurry lines from the ion exchange columns to the de-deuteration tank and from the de-deuteration tank to the resin disposal vault are shielded with 1 ft thick concrete.

The screens in the tanks can be removed for cleaning in the event they plug up. Flanged connections are provided for this purpose. The strainer screens have to be cleaned periodically.

All the make and break connections of the system are located under exhaust hoods to prevent tritium from escaping into the room.

The system is located in the D₂O management area and has been made semi-automatic in operation. Flexible hoses and quick disconnect couplings have been minimized in order to minimize D₂O leaks and the associated tritium hazard.

5.4.5.3 Design Bases

See Table 5.4-5.

5.4.5.4 Safety Evaluation

5.4.5.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Moderator Deuteration and De-Deuteration System:

- GDC-1 - Quality standards and records.
- GDC-2 - Design bases for protection against natural phenomena.
- GDC-4 - Environmental and missile design bases.

- GDC-50 - Containment design basis.
- GDC-51 - Fracture prevention of containment pressure boundary.
- GDC-53 - Provisions for containment testing and inspection.
- GDC-54 - Piping systems penetrating containment.
- GDC-56 - Primary containment isolation.

5.4.5.4.2 Assessment

Although the Moderate Deuteration and De-Deuteration System has no safety function, some components are located inside and some outside of containment and must be seismically qualified. Since the system has been classified as ASME Class 3, and with seismic requirements, the system must meet the requirements of GDC 1 for a Quality Assurance Program, GDC 2 for seismic analysis and qualification, and GDC 4 for missile protection.

The system has piping passing through the containment; therefore, it must meet the requirements of GDC 50, 51, 53, 54 and 56. Valves must be added to the piping that penetrates containment, one inside and one outside, and capable of automatic operation. In addition, the valves and piping that penetrates containment must meet the fracture toughness requirements of GDC 51.

5.4.5.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10CFR50. The containment isolation valves and the portions of piping that penetrate containment should be ASME Class 2.

- GDC 2 - The component supports components, and piping design should be verified that they meet the seismic requirements.
- GDC 4 - No change.
- GDC 50 - Piping penetrating containment should be ASME Class 2 and meet fracture toughness requirements of GDC 51. In addition, automatic operating isolation valves should be added to each line penetrating containment, one inside and one outside of containment. The valves should be ASME Class 2 and meet fracture toughness requirements of GDC 51.
- GDC 51 - See GDC 50 above.
- GDC 53 - No change.
- GDC 54 - See GDC 50 above.
- GDC 56 - See GDC 50 above.

5.4.6 Moderator D₂O Sampling System

5.4.6.1 Function

The purpose of the Moderator D₂O Sampling System is to evaluate the performance of various systems associated with the moderator.

5.4.6.2 Description

The Moderator D₂O Sampling System flow diagram is shown in Figure 5.4-6.

Samples of liquid may be obtained by hypodermic sampling, from the Main Moderator System, Moderator D₂O Collection System, Moderator Purification System and the D₂O Cleanup System. Samples may also be taken from the poison mixing tanks of the liquid poison system.

Control room indication of the Main Moderator System and Moderator Purification System conductivity is provided.

5.4.6.3 Design Bases

See Table 5.4.6.

5.4.6.4 Safety Evaluation

5.4.6.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Moderator D₂O Sampling System:

None.

5.4.6.4.2 Assessment

The Sampling System is not safety-related.

5.4.6.4.3 Recommended Design Change

None.

5.4.7 Primary Heat Transport Pressure and Inventory Control System

5.4.7.1 Function

The function of the pressure and inventory control system is to provide a means of pressure and inventory control for the pressurized closed D₂O loops of the primary heat transport system as well as provide adequate overpressure protection.

The basic design requirements of the system are as follows:

- a. To provide a means of controlling heat transport system pressure at the appropriate values over the full range of operating conditions.
- b. To limit pressure increases and pressure decreases to acceptable values.
- c. To accommodate main heat transport system coolant swells and shrinks associated with warmup, cooldown, power changes and various unit disturbances.
- d. To provide means of relatively quick heat transport system pressure recovery following sudden pressure power reductions.
- e. To provide a cooling flow of purified D₂O to the main heat transport pump glands.
- f. To provide adequate relief capacity for overpressure protection of the main heat transport system as well as a means to contain any relief from it.

- g. To provide a means of degassing the main heat transport system coolant.
- h. To provide a variable test pressure signal for use in testing the various protective systems employing pressure as a process parameter.

5.4.7.2 Description

The pressure and inventory control system flowsheet is shown in Figure 5.4-7. Basically, the system consists of a pressurizer, a degasser-condenser, and a feed and bleed circuit. The pressurizer is designed to accommodate the swell in the heat transport system from zero power hot to full power hot. This permits the reactor power to be increased or decreased at a rapid rate without imposing a severe requirement on the feed and bleed circuit. The feed and bleed circuit is designed to handle the shrink and swell rates which will take place during heatup and cooldown.

When the reactor is at power, pressure is controlled by the pressurizer with inventory being adjusted by the feed and bleed circuit. When the reactor is shutdown and the pressurizer isolated, pressure can be controlled by the feed and bleed circuit.

A. Pressurizer

The common pressurizer which controls the pressure in the two heat transport circuits is a cylindrical vessel installed vertically and is connected to lines linking the outlet headers at one end of the reactor. Figure 5.4-8 shows the pressurizer arrangement. The liquid D_2O stored in the pressurizer is heated by electric immersion heaters if its temperature falls below saturation

temperature. The cushioning effect of the D_2O steam volume in the pressurizer is supplemented by two 100% steam bleed valves which open when the pressure rises above a set point. The steam bleed valves discharge into the degasser-condenser. The pressurizer level set point is power level dependent. It increases to accommodate the swell between zero power hot and full power. The D_2O level is automatically controlled at the set point by the D_2O feed and bleed system under the control of the unit computers. The pressurizer is also equipped with a D_2O spray nozzle which permits cooling of the vessel walls for maintenance.

The pressurizer is designed and manufactured to the requirements for Class 1 Components in Section III of the ASME Boiler and Pressure Vessel Code.

B. Feed and Bleed Circuit

The feed and bleed circuit is provided primarily for inventory control of the heat transport system. Bleed flow is taken from the purification system return line as a cold and purified fluid. The bleed line runs to the D_2O storage tank which stores the swell from zero power cold to zero power hot. The feed flow is taken from the D_2O storage tank. Two high pressure multistage D_2O feed pumps are provided, one of which is normally operating with the other on standby. The feed flow is supplied to the main HTS via the interconnection at the purification system return line (see Figure 5.4-10). If the purification system is isolated for any reason, feed and bleed can still be maintained: feed via the

normal path and bleed via the degas line to the degasser-condenser, which is connected to the purification return line downstream of the isolating valves. The feed and bleed control valves are duplicated for reliability.

C. Overpressure Protection

Two instrumented relief valves provide overpressure protection for each of the two HTS main circuits during reactor shutdown, when the pressurizer may be isolated, and during abnormal transients in conjunction with the reactor safety systems. The pressure relief valves are connected to the lines from the outlet headers to the pressurizer on the header side of the pressurizer isolating valves (refer to Figure 5.4-7). The valves will discharge a mixture of steam and water into the degasser-condenser when they open.

D. Degasser - Condenser

A condenser is provided to receive steam bleed from the pressurizer and liquid relief and degassing flow directly from the heat transport system. The condensed D_2O is returned via a cooler to the storage tank and/or feed pumps. The D_2O level is controlled by level control valves located downstream of the cooler. Service water is used as a cooling agent. The flow of service water in tubes is regulated to control the pressure in the condenser to a value which is sufficient to transfer fluid at the maximum required flow rate into the storage tank. The heavy water from the degas line enters the degasser-condenser as a flashing spray which promotes degassing. Non-condensable gases can be vented to the off-gas management system.

E. Storage Tank

The storage tank is designed to accommodate the swell from the heat transport system from cold condition to the zero power hot condition. A helium cover gas blanket is maintained at approximately atmospheric pressures. Means are provided to vent the helium cover gas to the off-gases management system (refer to Figure 5.1-7).

F. Instrumentation and Control

Under normal power producing operation, the primary heat transport system pressure is controlled by the steam bleed valves and the heaters in the pressurizer. This pressure control system operates on a narrow range of pressure transmitter signal. Tripliated sets of pressure measurements taken from the outlet headers are used. The control system selects the outlet header that has the highest pressure and controls this pressure to the desired value. If the pressure is above the set point the steam bleed valves on the pressurizer open to reduce pressure and if below the set point a heater in the pressurizer tank will come on to raise pressure. There are six electric heaters in the bottom of the pressurizer, one variable and five fixed heaters.

A wide range pressure control system is used when the pressurizer is isolated. This is a single channel analog system and uses signals from pressure transmitters located on the outlet headers.

The highest of the outlet header pressures is selected and is compared to the set point in a pressure controller. The controller output is fed to the two feed valves and the two liquid

bleed valves which control the heat transport system pressure. The selection to wide range from narrow range control should not be used when the plant is at power since it does not have the reliability or capacity to provide adequate control. When the pressurizer is isolated in a standby condition, its pressure is controlled by a single channel pressure control loop which activates the steam bleed valves and the heaters. Overpressure protection of the pressurizer is provided by tripping open the steam bleed valves if the pressure becomes too high. Triplicated pressure measurements are used for this protection.

The D₂O level in the pressurizer is controlled by the feed and bleed valves which are governed by the unit computer. A median is selected from among three different level measurements and then compared to the programmed level set point. For high reliability the level control program can be run from either of the two unit computers.

Two separate single channel analog systems are used to control the D₂O level and the pressure in the degasser-condenser. The level control valve which is located downstream of the degasser cooler responds to signals from the level controller. The process water control valve is governed by the pressure controller.

Control of the degasser cooler outlet temperature is performed by a single channel analog system. The D₂O outlet temperature is limited because of the feed pump NPSH requirements. A high temperature protection is provided by an over-ride control which closes the degasser-condenser level control valve.

The degasser flow control is performed by adjusting the opening of the degasser flow control valve from a band controller on the panel. If the degasser-condenser pressure rises above its set point, by an arbitrary margin, the degasser valve is automatically closed down by an over-ride control to prevent the offgas system from being overpressurized.

5.4.7.3 Design Bases

Refer to Table 5.4-7.

5.4.7.4 Safety Evaluation

5.4.7.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Primary Heat Transport Pressure and Inventory Control System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 10 - Reactor design.
- GDC 11 - Reactor inherent protection.
- GDC 12 - Suppression of reactor power oscillations.
- GDC 13 - Instrumentation and control.
- GDC 14 - Reactor coolant pressure boundary.
- GDC 15 - Reactor coolant system design.
- GDC 30 - Quality of reactor coolant pressure boundary.
- GDC 31 - Fracture prevention of reactor coolant pressure boundary.
- GDC 32 - Inspection of reactor coolant pressure boundary.

5.4.7.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied within the design and construction of the Primary Heat Pressure Transport and Inventory Control System.

The components of the system appear to meet the requirements of GDC 2, but would require verification that the seismic loads are included in the analysis.

A missile protection evaluation would have to be performed to verify that all components are amply protected from the effect of internally generated missiles. This evaluation is required by GDC 4.

The requirements of GDC 10 through 15 inclusive, are not fully satisfied with this present design. The Heat Transfer Pressure and Inventory Control System possesses suitable capability in equipment to perform its function. However, since the pressurizer must be isolated in the event of a loss-of-coolant-accident, it is judged that the two isolation valves located on the lines connecting the pressurizer with the outlet headers at one end of the reactor do not provide enough reliability for isolating the pressurizer from the heat transport system. The pressurizer surge line is connected in-between these two valves. Assuming that when the loss-of-coolant logic is actuated at least one of these valves close, the two PHT loops are isolated from one another. However, if the valve located near the reactor outlet header of the ruptured loop fails to close then the pressurizer D₂O inventory will be discharged into the ruptured loop and ultimately to the

containment atmosphere. This can be corrected by installing an isolation valve in the pressurizer surge line, in effect duplicating each of the two existing pressurizer isolation valves.

GDC 30 and 31 will require that the components of the system be constructed from material that will meet the fracture toughness requirements of the criterion and for ASME Class 1 components.

An inservice inspection program is required for the Primary Heat Transport Pressure and Inventory Control System that is in accordance with ASME Section XI and 10 CFR 50.55a. This will require providing access to areas requiring inspection in the layout of the plant. Also, all welds will have to be finished to permit a baseline inspection and periodic inservice inspection. Removable insulation will be required for those areas to be inspected.

5.4.7.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Evaluate the design and analyses of all components of the system to assure that the seismic forces have been accounted for.
- GDC 4 - The components should be provided with pipe-whip restraints, where required, to prevent other piping or components from damage.

- GDC 10 - Provide an additional automatic operating isolation valve in the pressurizer surge line.
- GDC 11 - See GDC 10, above.
- GDC 12 - See GDC 10, above.
- GDC 13 - See GDC 10, above.
- GDC 14 - See GDC 10, above.
- GDC 15 - See GDC 10, above.
- GDC 30 - The material of construction of the pressure boundary components should be selected to provide an adequate margin between its nil-ductility transition temperature and the minimum temperature at which the components could be partially pressurized.
- GDC 31
- GDC 32 - Provide an inservice inspection program that provides for the examination of every weld and mechanical connection in the pressure boundary components and assembly.

5.4.8 Gland Seal System

5.4.8.1 Function

The function of the gland seal system is to supply cooled and filtered D₂O at high pressure to the primary heat transport pump glands.

5.4.8.2 Description

The gland seal system flow diagram is shown in Figure 5.4-9. An adequate amount of D₂O is pumped by the D₂O feed pumps through the gland heat exchanger and filters and supplied to the four heat transport pump glands at about 100°F and at high pressure. Part of this flow passes through the heat transport pump throttle brushing into the Heat Transport System while the balance flows up past the pump seals and is collected at low pressure and returned to the D₂O storage tank.

Additional connections to two heat transport pump Discharge lines supply high pressure D₂O to the gland seal system in the event that the D₂O fuel pumps are unavailable. The gland seal system heat exchanger is sized to cool D₂O at the reactor inlet temperature down to 100°F. Two disposable paper cartridge filters are connected in parallel, downstream of the heat exchanger. Flow, temperature, and pressures are monitored (and alarmed if necessary) at various points in the system.

Low flow, measured in the common supply line to the pumps, will automatically close the gland return flow isolating valve. This will bottle up the system and prevent outflow of hot water from the heat transport system that could damage the pump seals. The reliability of the system is high since two feed pumps are available for normal supply and back up is available from the discharge side of two main pumps. The plant computers are used for the alarms and logic in the gland seal system.

5.4.8.3 Design Bases

Refer to Table 5.4-8.

5.4.8.4 Safety Evaluation

5.4.8.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Gland Seal System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 50 - Containment design basis.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 53 - Provisions for containment testing and inspection.
- GDC 54 - Piping systems penetrating containment.
- GDC 56 - Primary containment isolation.
- GDC 57 - Closed system isolation valves.

5.4.8.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied within the design and construction of the Gland Seal System. In addition, the shell side of the Gland Seal Cooler should be reclassified from ASME Section VIII to ASME Section III, Class 2 or Class 3.

The components of the system appears to meet the requirements of GDC 2, but would require verification that the seismic loads are included in the analysis.

A missile protection evaluation would have to be performed to verify that all components are amply protected from the effect of internally generated missiles. This evaluation is required by GDC 4.

GDC 50, 51, 53, 54, 56, and 57 provide criteria for piping that penetrate the containment and for assuring that containment isolation can be achieved. The provisions of these criteria are not fully complied with in the present design of the PHWR. Each line that penetrates the containment must be designed to either ASME Class 1 or 2 and shall meet the fracture toughness of the pipe material required by GDC 51. In addition, each line must be provided with an isolation valve inside the containment and one outside the containment.

5.4.8.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.

The shell of the Gland Seal Cooler should be ASME Class 2 or 3 and all piping penetrating the containment should be ASME Class 1 or 2.

GDC 2 - Evaluate the design and analysis of all components of the system to assure that the seismic forces have been accounted for.

GDC 4 - The high-pressure side of the system should be provided with pipe-whip restraints, where required, to prevent other piping or components from damage.

GDC 50 - Piping penetrating containment should have fracture-toughness properties to meet the requirements of this criterion. Each line requires containment isolation valves which shall also meet the fracture-toughness property requirements.

GDC 51 - See GDC 50, above.

GDC 53 - No change.

GDC 54 - See GDC 50, above.

GDC 56 - See GDC 50, above.

GDC 57 - See GDC 50, above.

5.4.9 Primary Heat Transport Purification System

5.4.9.1 Function

The purification system limits the activity and corrosion product buildup in the reactor coolant by removing soluble and insoluble impurities and by maintaining the pH of the D₂O at the required value. It can also remove crud at a high rate following a crud burst caused by a chemical, hydraulic or temperature transient. The maximum purification flow rate is 5.6×10^5 lb/hr at a temperature of 150°F. This flow rate gives a purification half life of about 20 minutes. The time taken to reduce the concentration of dissolved solids by 50%, assuming no addition of solids, is defined as purification half life.

5.4.9.2 Description

One purification system, common to both heat transport loops, is provided. The purification system flow diagram is shown in Figure 5.4-10. Flow is taken from one reactor inlet header of each heat transport loop via the shutdown cooling piping, passed through an interchanger, cooler, filters and returned to one pump suction in each circuit. The interchanger - cooler combination is used to first cool the heavy water and then recover heat to minimize heat losses. When the flow is at its maximum value, the cooler outlet temperature is 150°F with an interchanger return temperature of 455°F. The feed and bleed system is connected to the purification system.

A control valve located downstream of the purification ion exchange columns is provided for purification flow control. The purification system can be isolated for maintenance purposes by means of isolation lines located in the inlet and outlet lines of the system. In the event of a loss of coolant accident, the purification system will be isolated automatically.

A. Filters

Two disposable paper cartridge filters connected in parallel are provided with a rated flow rate of 2.8×10^5 lb/hr each. Normally both filters are used but the cartridges in a filter can be changed while still maintaining flow through the other.

B. Ion Exchange Columns

Two ion exchange columns are provided, each rated at 2.8×10^5 lb/hr. The systems are designed so that the resin can be transferred in and out of one ion exchange column while the other is maintained in operation. Each column contains about 38 ft^3 of resin and has a total internal volume of about 50.5 ft^3 .

C. Interchanger/Cooler

A heat interchanger and cooler combination is used to cool down the heat transport coolant from reactor inlet temperature to the justification temperature of 150°F , and to recover about 90% of the heat. The materials used for tubing in the heat exchanger are carbon steel for the interchanger and Incoloy-800 alloy for the cooler.

D. Instrumentation and Control

Instrumentation is required to monitor temperature, flows, and pressures in the system as well as the pressure drops across the filters and ion exchangers.

Temperature control on the outlet of the purification cooler is performed by a single channel analog system. The temperature

control valve responds to signals from the controller to modify the flow rate of high pressure recirculated water. To prevent damage to the resin in the ion exchange columns, a temperature over-ride is provided to close the purification control valve if the temperature downstream of the cooler exceeds the set point by a predetermined margin.

The four isolation valves of the purification system are normally open and closed only when the loss of coolant logic is actuated. If these valves are actuated then the pressurizer isolating valves are also closed and two primary heat transport circuits are isolated from each other. Although the ability to control the system pressure is lost, overpressure is still provided on the heat transport system and the pressurizer.

5.4.9.3 Design Bases

Refer to Table 5.4-9.

5.4.9.4 Safety Evaluation

5.4.9.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Primary Heat Transport Purification System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 14 - Reactor coolant pressure boundary.
- GDC 15 - Reactor coolant system design.
- GDC 30 - Quality of reactor coolant pressure boundary.
- GDC 31 - Fracture prevention of reactor coolant pressure boundary.

GDC 32 - Inspection of reactor coolant pressure boundary.

5.4.9.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied within the design and construction of the Primary Heat Transport Purification System. The shell side of the Heat Transport Purification Cooler should be reclassified from ASME Section VIII to ASME Section III, Class 2 or Class 3.

The components of the system appears to meet the requirements of GDC 2, but would require verification that the seismic loads are included in the Analysis.

A missile protection evaluation would have to be performed to verify that all components are amply protected from the effects of internally generated missiles. This evaluation is required by GDC 4. The high-pressure pipe lines would have to be provided with pipe-whip restraints, where required, to prevent damage to other piping systems and safety-related components.

GDC 14 and 15 appears acceptable from a licensing viewpoint.

GDC 30 and 31 will require that all material used in the pressure boundary components meet the fracture toughness requirements of GDC 31 and the ASME Boiler & Pressure Vessel Code, Section III.

The components of this system should be layed out to provide access for performing inservice inspection. An inservice inspection program will be required for these components to comply with the requirements of GDC 32, 10 CFR 50.55a, and ASME Section XI.

5.4.9.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
The shell side of the Heat Transport Purification Cooler should be Class 2 or Class 3.
- GDC 2 - Evaluate the design and analysis of all components of the system to assure that the seismic forces have been accounted for.
- GDC 4 - The high-pressure side of the system should be provided with pipe-whip restraints, where required, to prevent other piping or components from damage.
- GDC 14 - No change.
- GDC 15 - No change.
- GDC 30 - The material for the pressure boundary components should be
GDC 31 selected to provide an adequate margin between its nil-ductility transition temperature and the minimum temperature at which the components will be partially pressurized.
- GDC 32 - Provide an inservice inspection program that provides for the examination of every weld and mechanical connection in the pressure boundary components. In addition, the ISI program should provide for functional testing of valves and pumps.

5.4.10 Primary Heat Transport Deuteration and De-Deuteration System

5.4.10.1 Function

The function of the PHT deuteration and de-deuteration system is to replace the light water contained in the fresh ion exchange resin, by heavy water (deuteration) and to replace the heavy water containment in the resin which was used in the PHT purification system by light water (de-deuteration).

Deuteration is the replacement of H_2O in the resin by D_2O by a slow upward flow of D_2O through the resin bed. De-deuteration is the replacement of D_2O in the resin by H_2O by a slow downward flow of H_2O through the resin bed.

Some heavy water is downgraded during both the deuteration and the de-deuteration process. This water will be collected and upgraded for further use.

5.4.10.2 Description

The heat transport deuteration and de-deuteration system is shown in Figure 5.4-11. Besides valves, strainers, instruments, piping, flexible bases and fittings necessary to complete the system, the following equipment is used: one deuteration tank, one de-deuteration tank, one canned pump, one D_2O tank, and one H_2O tank.

The deuteration and de-deuteration tanks are about 4 feet in diameter and about 6 feet tall and hold approximately 38 ft^3 of ion exchange resin. Both tanks have a conical bottom to facilitate resin removal by resin slurry. They are provided with removable 'wedge wire' screens capable of withstanding high differential pressure.

Both the D_2O tank and the H_2O tank have a capacity of about 81 ft^3 each and are provided with multiple position level switches and alarms. The level switches (4 for the D_2O tank and 5 for the H_2O tank) operate the automatic ON-OFF control valves located in the deuteration tank outlets. The pump is used for D_2O service only and has a capacity of about 75 gpm at 100°F . All the strainers in the system as well as the wedge wire screens retain any particles larger than 0.2 mm. The resin bead size is 0.38-0.60 mm.

All the valves in the system are manual valves, except the two located in the main lines to the storage tanks in the D_2O cleanup system. These two valves will be ON-OFF automatically operated control valves. Diaphragm valves are used for D_2O service, and ball valves where resin transfer is involved.

The deuteration and de-deuteration processes can be carried on at the same time, independently of each other. When the deuteration and the de-deuteration processes are completed, the deuteration tank will be full of heavy water and the de-deuteration tank full of light water. The H_2O in the de-deuteration tank will be transferred to the resin disposal vault and then the D_2O from the deuteration tank will be transferred to the de-deuteration tank. These two transfers are made using instrument air.

To eliminate inadvertent downgrading of D_2O , flexible connections are provided wherever it is possible to mix D_2O with H_2O , that is, an operator will have to make the connection before D_2O and H_2O are mixed.

The deuteration and de-deuteration system is located in the Reactor Building. Only the de-deuteration tank is shielded with five foot thick light

concrete. One shielding panel can be removed to allow access to the vessel. All resin slurry lines are shielded with one foot thick concrete.

All the make and break connections of the system will be located under exhaust hoods to prevent tritium from escaping into the room. The sampling and D₂O collection parts of the system are semi-automatic and physically located in the D₂O management area.

The deuteration and de-deuteration system is provided with the following instrumentation:

Deuteration - local and control room indications of the D₂O level in D₂O tank and local indications of the specific gravity of downgraded D₂O and of the flow of downgraded D₂O to D₂O clean-up system tanks.

De-deuteration - local and control room indications of H₂O level in the H₂O tank and local indications of the specific gravity of downgraded D₂O and of the flow of downgraded D₂O to D₂O clean-up system tanks.

High levels in the D₂O tank and H₂O tank are both annunciated. Level measurement is interlocked with automatic valve to automatically close it at pre-set tank levels. These tank levels are related to the isotopic of the downgraded D₂O leaving the fresh resin tank.

5.4.10.3 Design Bases

Refer to Table 5.4-10.

5.4.10.4 Safety Evaluation

5.4.10.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Primary Heat Transport Deuteration and De-Deuteration System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.

5.4.10.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with in the design and construction of the Primary Heat Transport Deuteration and De-Deuteration System.

The components of the system appears to meet the requirements of GDC 2, but would require verification that the seismic loads are included in the analysis.

The design of the system should be evaluated for possible missile impact damage and provisions should be made to protect vital equipment.

5.4.10.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Evaluate the design and analysis of all components of the system to assure that the seismic forces have been accounted for.
- GDC 4 - Evaluate the design to assure that vital components of the system are protected from the impact of missiles.

5.4.11 Primary Heat Transport D₂O Sampling System

5.4.11.1 Function

The D₂O sampling system is used to obtain samples of heavy water from various points in the primary heat transport system.

5.4.11.2 Description

One sample cabinet is used for this system. Samples of liquid may be obtained by hypodermic sampling, from the heat transport purification system, the gland seal cooling system, the D₂O collection tank, the discharge lines of heat transport pumps No. 2 and 4, and heat transport inlet lines to boilers No. 2 and 4. A process flow diagram of the PHT D₂O sampling system is shown in Figure 5.4-12.

Crud filters with flow integrators are provided for measuring the solid impurities in the heat transport system. Hypodermic and canister samples can be withdrawn for further analysis. Sample coolers and pressure reducing coils are provided in the lines connecting the sample cabinet with the heat transport system. The heavy water exiting the crud filters is returned to the D₂O clean-up system. The sample cabinet is connected to the heat transport system feed pump suction line to close the circuit and return the unused heavy water. Any spills collect in the cabinet which is fitted with a moisture detector.

5.4.11.3 Design Bases

Refer to Table 5.4-11.

5.4.11.4 Safety Evaluation

5.4.11.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Primary Heat Transport D₂O Sampling System:

None.

5.4.11.4.2 Assessment

The Sampling System is not safety-related.

5.4.11.4.3 Recommended Design Change

None.

5.4.12 Primary Heat Transport D₂O Collection System

5.4.12.1 Function

The D₂O collection system is provided to collect leakage from mechanical components, and to receive D₂O drained from equipment prior to maintenance. The collected D₂O is transferred to the D₂O storage tank or, if the D₂O is downgraded, to drums for transport to the upgrading column.

5.4.12.2 Description

The flow diagram of the heat transport D₂O collection system is shown in Figure 5.4-13. The collection tank is located at a low elevation relative to the heat transport equipment, and pumps are provided to transfer D₂O from the collection tank to the storage tank.

The D₂O collection tank collects any water that has leaked from the components of the heat transport system. Two pumps and a common discharge valve are associated with the tank. Computer logic is arranged to start the pumps and open the valve on high level in the collection tank. The D₂O collected is then pumped back to the storage tank. At low level, the pumps are automatically turned off and the isolating valve closes. Samples of the contents of the tank can be delivered to the PHT D₂O Sampling System for isotopic analysis. If the isotopic purity is low then the collection tank contents are pumped to drums for upgrading. Level and temperature of the tank contents can be indicated and alarmed by the unit computer.

The cover gas in the collection tank is helium. The helium pressure is maintained at the atmospheric level by means of a connection to the heat transport D₂O storage tank which is at atmospheric pressure.

The D₂O collection system components are constructed from austenitic stainless steel to avoid corrosion.

5.4.12.3 Design Bases

Refer to Table 5.4-12.

5.4.12.4 Safety Evaluation

5.4.12.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Primary Heat Transport D₂O Collection System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.

5.4.12.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with in the design and construction of the Collection System.

The components of the system appear to meet the requirements of GDC 2, but would require verification that the seismic loads are included in the analysis.

The design of the system should be evaluated for possible missile impact damage and provisions should be made to protect vital equipment.

5.4.12.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Evaluate the design and analysis of all components of the system to assure that the seismic forces have been accounted for.
- GDC 4 - Evaluate the design to assure that vital components of the system are protected from the impact of missiles.

TABLE 5.4-1

MAIN MODERATOR SYSTEM DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
Main Circuit	
General:	
Moderator	Heavy Water
Calandria Inlet Temperature	110°F
Calandria Outlet Temperature	160°F
Moderator Flow Rate	15,000 gpm
Cooling Capacity	120 MW _t
Applicable Code	ASME Section III, Class 3
Heat Exchangers	
Quantity	2
Type	Vertical U-tube in shell
Duty	Continuous
Capacity	50%
Heat Transferred	60 MW _t
Overall Length	28 ft
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	7,500 gpm
Inlet Temperature	160°F
Outlet Temperature	110°F
Design Pressure	240 psig
Design Temperature	200°F

TABLE 5.4-1 (cont'd)

MAIN MODERATOR SYSTEM DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	14,000 gpm
Inlet Temperature	65°F
Outlet Temperature	94°F
Design Pressure	150 psig
Design Temperature	150°F
Construction	
Tube Material	Titanium
Tube Outside Diameter	0.625 in
Shell Material	--
Applicable Codes	
Tube Side	ASME-Section III, Class 3
Shell Side	ASME-Section VIII
Pumps	
Quantity	2
Type	Vertically mounted double suction centrifugal
Duty	Continuous
Capacity	100%
Pump Data:	
Fluid	D ₂ O
Flow Rate	15,000 gpm

TABLE 5.4-1 (cont'd)

MAIN MODERATOR SYSTEM DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
Operating Temperature:	160°F
Head	180 ft
Design Pressure	240 psig
Design Temperature	200°F
Main Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Power Supply	Class IV
Pony Motor Data:	
Type	--
Speed	½ Main Motor Speed
Power Supply	Class III
Construction:	
Materials in Contact with D ₂ O	Austenitic Stainless Steel
Applicable Code	ASME-Section III, Class 3
Tanks	
Name	Calandria Head Tank
Quantity	1
Type	Horizontal cylinder with dished heads
Capacity	100 ft ³
Design Conditions:	
Operating Pressure	0 psig

TABLE 5.4-1 (cont'd)

MAIN MODERATOR SYSTEM DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
Operating Temperature	110 F
Design Pressure	15 psig
Design Temperature	250 F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME-Section III-Class 3

TABLE 5.4.2

Moderator Purification System

Design Parameters

<u>Item</u>	<u>Value</u>
General	
Purification Flow Rates:	125 gpm
Normal	250 gpm
Maximum	110°F
Ion Exchange Inlet Temperature	
Cleanup Half Lives:	6 hours
At Normal Flow	3 hours
At Maximum Flow	ASME Section III, Class 3
Applicable Code	
Heat Exchangers	
Name	Moderator Purification Heat Exchanger
Quantity	1
Type	Tube in Shell
Duty	Continuous
Capacity	100%
Heat Transferred	2 MW
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	250 gpm
Inlet Temperature	160°F

TABLE 5.4.2 (cont'd)

Moderator Purification System

Design Parameters

<u>Item</u>	<u>Value</u>
Outlet Temperature	110°F
Design Pressure	240 psig
Design Temperature	200°F
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	410 gpm
Inlet Temperature	75°F
Outlet Temperature	112°F
Design Pressure	150 psig
Design Temperature	200°F
Construction:	
Tube Material	Incoloy
Shell Material	Carbon Steel
Applicable Codes:	
Tube Side	ASME Section III, Class 3
Shell Side	ASME Section VIII
Filters and Strainers	
Name	Purification Filter
Quantity	1

TABLE 5.4.2 (cont'd)

Moderator Purification System

Design Parameters

<u>Item</u>	<u>Value</u>
Type	Removable Cartridges housed in Pressure Vessel
Filter Porosity	5-10 micron
Clean P at Normal Flow	1 psig
Dirty P at Normal Flow	30 psig
Design Conditions:	
Normal Flow Rate	125 gpm
Maximum Rated Flow	250 gpm
Cartridge Material	Pleated Paper
Operating Pressure (Gauge)	90 psig
Operating Temperature	110°F
Design Pressure (Gauge)	240 psig
Design Temperature	200°F
Construction:	
Material	Stainless Steel
Applicable Code	ASME Section III, Class 3
Ion Exchangers	
Name	Moderator Ion Exchange Columns
Quantity	5
Type	Vertical Cylinder with Dished Top and Conical Bottom

TABLE 5.4.2 (cont'd)

Moderator Purification System

Design Parameters

<u>Item</u>	<u>Value</u>
Volume	9 ft ³
Resin Capacity	7 ft ³
Type of Screen	Wedge-Wire
Design Conditions:	
Maximum Flow Rate	125 gpm
Direction of Flow	Downward
Operating Pressure	90 psig
Operating Temperature	110°F
Design Pressure 240 psig	
Design Temperature	200°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 5.4.3

Moderator Cover Gas System

Design Parameters

<u>Item</u>	<u>Value</u>
General:	
Cover Gas	Helium
Cover Gas Flow Rate	4 ft ³ /min
Applicable Code	ASME Section III, Class 3
Heat Exchangers	
Name	Cover Gas Heater
Quantity	1
Type	Electric Heating Element enclosed in a Shell
Duty	Continuous
Heat Transferred	0.5 kW
Shell Side Data:	
Fluid	Helium and D ₂ O Vapor
Flow Rate	4 ft ³ /min
Inlet Temperature	160°F
Outlet Temperature	360°F
Design Pressure	15 psig
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 5.4.3 (cont'd)

Moderator Cover Gas System

Design Parameters

<u>Item</u>	<u>Value</u>
Compressors	
Name	Cover Gas Compressors
Quantity	2
Type	Diaphragm
Duty	Continuous
Capacity	100%
Compressor Data:	
Fluid	Helium and D ₂ O Vapor
Flow Rate	4 ft ³ /min
Operating Temperature	160°F
Head	2 psig
Design Pressure	15 psig
Design Temperature	250°F
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Power Supply	Class III
Construction:	
Materials in Contact with Cover Gas	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Recombination Equipment	

TABLE 5.4.3 (cont'd)

Moderator Cover Gas System

Design Parameters

<u>Item</u>	<u>Value</u>
Name	Recombination Units
Quantity	2
Type	Palladium Catalyst enclosed in a Vertical Cylinder with Dished Heads
Capacity	100%
Design Conditions:	
Operating Pressure	2 psig
Operating Temperature	360°F
Design Pressure	15 psig
Design Temperature	800°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 5.4.4

Moderator D₂O Collection System

Design Parameters

<u>Item</u>	<u>Value</u>
General:	
Expected Collection Flow Rate	1.3 gpm
Return Flow Capacity	18 gpm
Pumps	
Name	D ₂ O Collection Pump
Quantity	1
Type	Canned Centrifugal
Duty	Intermittant
Pump Data:	
Fluid	D ₂ O
Flow Rate	18 gpm
Operating Temperature	100 to 200°F
Head	100 ft
Design Pressure	150 psig
Design Temperature	210°F
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Power Supply	Class IV

TABLE 5.4.4 (cont'd)

Moderator D₂O Collection System

Design Parameters

<u>Item</u>	<u>Value</u>
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Tanks	
Name	D ₂ O Collection Tank
Quantity	1
Type	Horizontal Cylinder with Dished Heads
Capacity	11 ft ³
Design Conditions:	
Operating Pressure	2 psig
Operating Temperature	100 to 200°F
Design Pressure	30 psig
Design Temperature	210°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 5.4.5

Moderator Deuteration and De-Deuteration System

Design Parameters

<u>Item</u>	<u>Value</u>
General:	
Flow Rate	1.3 to 2.5 gpm
D ₂ O Required to Deuterate 7 ft of Resin	9.5 ft ³
D ₂ O/H ₂ O Interface Velocity	0.01 to 0.02 ft/min
Applicable Code	ASME Section III, Class 3
Pump	
Name	Resin Transfer Pump
Quantity	1
Type	Centrifugal
Duty	Intermittant
Pump Data:	
Fluid	D ₂ O
Flow Rate	25 gpm
Operating Temperature	Ambient
Head	210 ft
Design Pressure	150 psig
Design Temperature	150°F
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Power Supply	Class IV

TABLE 5.4.5 (cont'd)

Moderator Deuteration and De-Deuteration System

Design Parameters

<u>Item</u>	<u>Value</u>
Construction:	
Materials in Contact with D ₂ O	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Tanks	
Name	D ₂ O and H ₂ O Head Tanks
Quantity	2 (1 for each service)
Type	Vertical Cylinder with Dished Heads
Capacity	10.9 ft ³
Design Conditions:	
Operating Pressure	Atmospheric
Operating Temperature	100°F
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Materials	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Name	Resin Deuteration and De-deuteration Tanks
Quantity	2 (1 for each service)
Type	Vertical cylinder with dished top and conical bottom

TABLE 5.4.5 (cont'd)

Moderator Deuteration and De-Deuteration System

Design Parameters

<u>Item</u>	<u>Value</u>
Capacity	9 ft ³
Design Conditions:	
Operating Pressure	0 to 150 psig
Operating Temperature	Ambient
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 5.4.6

Moderator D₂O Sampling System

Design Parameters

<u>Item</u>	<u>Value</u>
General:	
Type of Sample	Hypodermic
Points Sampled	Main moderator system Purification system return Individual ion exchanger outlets Moderator D ₂ O collection system D ₂ O cleanup system
Sample Pots:	
Number	6
Type	Hypodermic
Material:	
Body	Austenitic Stainless Steel
Septum	Neoprene
Conductivity Cell Holders:	
Number	2
Purpose	To contain in-line conductivity cells for continuous monitoring of moderator conductivity.
Conductivity Range	0 to 5 micromho/cm
Applicable Code	ASME Section III, Class 3

TABLE 5.4-7

Primary Heat Transport Pressure and Inventory Control System

Design Data

<u>Item</u>	<u>Value</u>
Pressurizer	
Quantity	1
Type	Vertical cylinder with electrical immersion heaters
Capacity	1600 ft ³
Design Conditions:	
Operating Pressure	1449 psig
Operating Temperature	590°F
Design Pressure	1600 psig
Design Temperature	650°F
Construction:	
Material	Carbon steel
Applicable Code	ASME Section III, Class 1
Immersion Heaters	
Duty	Intermittent
Capacity	1 MW(e)
Degasser-Condenser	
Quantity	1
Type	Vertical cylinder with inverted U-tube cooler
Duty	Intermittent
Heat Transferred	15.55 MW(th)

TABLE 5.4-7 (cont'd)

Primary Heat Transport Pressure and Inventory Control System

Design Data

<u>Item</u>	<u>Value</u>
Tube Side Data:	
Fluid	H ₂ O
Flow Rate	1750 gpm
Inlet Temperature	77°F
Outlet Temperature	144°F
Design Pressure	150 psig
Design Temperature	200°F
Shell Side Data:	
Fluid	D ₂ O (Flashing)
Flow Rate	1.5 x 10 ⁵ lb/hr
Inlet Temperature	293°F
Inlet Enthalpy	607.6 BTU/lb
Outlet Temperature	293°F
Outlet Enthalpy	253.8 BTU/lb
Design Pressure	1550 psig
Design Temperature	600°F
Construction:	
Tube Material	Incoloy 800
Tube Outside Diameter	0.625 inch
Shell Material	Carbon Steel
Applicable Codes	
Channel	ASME Section VIII
Shell and tubes side	ASME Section III, Class 1

TABLE 5.4-7 (cont'd)

Primary Heat Transport Pressure and Inventory Control SystemDesign Data

<u>Item</u>	<u>Value</u>
Pressure and Inventory Control Cooler	
Quantity	1
Type	Horizontal U-tube in shell
Duty	Intermittent
Capacity	100°C
Heat Transferred	6.3 MW(th)
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	1.5 x 10 ⁵ lb/hr
Inlet Temperature	293°F
Outlet Temperature	150°F
Design Pressure	1550 psig
Design Temperature	600°F
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	1125 gpm
Inlet Temperature	77°F
Outlet Temperature	118°F
Design Pressure	150 psig
Design Temperature	200°F

TABLE 5.4-7 (cont'd)

Primary Heat Transport Pressure and Inventory Control System

Design Data

<u>Item</u>	<u>Value</u>
Construction	
Tube Material	Incoloy 800
Tube Outside Diameter	16 mm 0.625 inch
Shell Material	Carbon Steel
Applicable Codes	
Tube Side	ASME Section, Class 1
Shell Side	ASME Section VIII
D ₂ O Feed and Bleed Circuit	
Function	
Bleed:	
Design Flowrate	275 gpm
Normal Flowrate	
Temperature	150°F
Feed:	
Design Flowrate	275 gpm
Normal Flowrate	
Temperature	150°F
Design Heat Transport Warmup and Cooldown rate	5°F/min
D ₂ O Feed Pumps	
Quantity	2

TABLE 5.4-7 (cont'd)

Primary Heat Transport Pressure and Inventory Control System

Design Data

<u>Item</u>	<u>Value</u>
Type	Multistage centrifugal
Duty	One continuous One on standby
Capacity	100%
Pump Data:	
Fluid	D ₂ O
Flow Rate	137.5 gpm
Temperature	150°F
Head	3700 ft
Design Pressure	2200 psig
Design Temperature	220°F
Motor Data:	
Type	Horizontal, squirrel cage, induction motor
Power Supply	Class III, 186 KW (250 HP)
Construction	
Materials in contact with fluid	
Applicable Code	ASME Section III, Class 2
D ₂ O Storage Tank	
Quantity	1
Type	Horizontal Cylinder
Capacity	1400 ft ³

TABLE 5.4-7 (cont'd)

Primary Heat Transport Pressure and Inventory Control System

Design Data

<u>Item</u>	<u>Value</u>
Design Conditions:	
Operating Pressure	Atmospheric
Operating Temperature	100°F
Design Pressure	15 psig
Design Temperature	220°F
Construction:	
Material	Carbon Steel
Applicable Code	ASME Section III, Class 2

TABLE 5.4-8

Gland Seal System Design Data

<u>Item</u>	<u>Value</u>
Gland Seal Cooler	
Quantity	1
Type	U-Tube in steel
Duty	Continuous
Capacity	100%
Heat Transferred	1.75 MW(th)
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	1.5 x 10 ⁶ lb/hr
Inlet Temperature	512°F
Outlet Temperature	115°F
Design Pressure	2,200 psig
Design Temperature	535°F
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	562.5 gpm
Inlet Temperature	77°F
Outlet Temperature	99°F
Design Pressure	150 psig
Design Temperature	200°F
Construction:	
Tube Material	Incoloy 800
Tube Outside Diameter	.625 inch
Shell Material	Carbon Steel

TABLE 5.4-8 (cont'd)

Gland Seal System Design Data

<u>Item</u>	<u>Value</u>
Applicable Codes	
Tube Side	ASME Section III, Class 1
Shell Side	ASME Section VIII
Filters	
Quantity	2
Type	Permanent vessel, disposable cartridges
Design Conditions:	
Design Flow	75 gpm
Operating Flow	27.5 gpm
Screen Size	1.2 micron
Design Pressure	2,200 psig
Design Temperature	535°F
Construction:	
Material	Carbon steel (shell)
Applicable Code	ASME Section III, Class 1

TABLE 5.4-9

Primary Heat Transport Purification System Design Data

<u>Item</u>	<u>Value</u>
Heat Transport Purification Circuit	
Purification Flow Rate	5.6×10^5 lb/hr
Ion Exchange Inlet Temperature	150°F
Clean-up Half-life	Approximately 20 minutes
Applicable Code	ASME Section III, Class 1

Heat Transport Purification Interchanger

Quantity	1
Type	Tube in shell
Duty	Continuous
Capacity	100%
Heat Transferred (normal)	50 MW(th)

Overall Length

Tube Side Data:

Fluid	D ₂ O
Flow Rate	5.6×10^5 lb/hr
Inlet Temperature	512°F
Outlet Temperature	210°F
Design Pressure	1870 psig
Design Temperature	535°F

Shell Side Data:

Fluid	D ₂ O
Flow Rate	5.6×10^5 lb/hr

TABLE 5.4-9 (cont'd)

Primary Heat Transport Purification System Design Data

<u>Item</u>	<u>Value</u>
Inlet Temperature	150°F
Outlet Temperature	455°F
Design Pressure	1807 psig
Design Temperature	535°F
Construction:	
Tube Material	
Tube Outside Diameter	5/8 inch
Shell Material	Carbon Steel
Applicable Codes:	
Tube Side	ASME Section III, Class 1
Shell Side	ASME Section III, Class 1
Heat Transport Purification Cooler	
Quantity	1
Type	Tube-in-shell
Duty	Continuous
Capacity	100%
Heat Transferred	10 MW(th)
Overall Length	
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	5.6 x 10 ⁵ lb/hr

TABLE 5.4-9 (cont'd)

Primary Heat Transport Purification System Design Data

<u>Item</u>	<u>Value</u>
Inlet Temperature	210°F
Outlet Temperature	150°F
Design Pressure	1870 psig
Design Temperature	535°F
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	3750 gpm
Inlet Temperature	65°F
Outlet Temperature	85°F
Design Pressure	150 psig
Design Temperature	200°F
Construction:	
Tube material	Incoloy 800
Tube outside diameter	5/8 inch
Shell Material	carbon steel
Applicable Codes	
Tube Side	ASME Section III, Class 1
Shell Side	ASME Section VIII
Heat Transport Purification Filters	
Quantity	2
Type	Removable cartridges housed in pressure vessel

TABLE 5.4-9 (cont'd)

Primary Heat Transport Purification System Design Data

<u>Item</u>	<u>Value</u>
Filter porosity	5 micron, nominal
Crud holding capacity	each approx. 60 lb
Clean P at normal flow	1 psig
Dirty P at normal flow	30 psig
Design conditions	
Normal flow rate each	562.5 gpm
Maximum flow rate, each	1125 gpm
Direction of flow in cartridges	Radially inward
No. of cartridges per filter	
Cartridge material	Pleated paper
Operating Pressure	1500 psig
Operating Temperature	150°F
Design Temperature filter vessel	535°F
Construction:	
Material	Carbon steel
Applicable Code	ASME Section III, Class 1
Heat Transport Purification Ion Exchange Columns	
Quantity	2
Type	Vertical cylinder, dished top, conical bottom
Volume	50.5 ft ³

TABLE 5.4-9 (cont'd)

Primary Heat Transport Purification System Design Data

<u>Item</u>	<u>Value</u>
Resin Capacity	38 ft ³
Type of Screen	wedge-wire, 0.008 in slot
Design Conditions:	
Maximum flow rate, each	562.5 gpm
Direction of flow	downward
Operating pressure	1500 psig
Operating temperature	150°F
Design Pressure	1870 psig
Design Temperature	535°F
Construction:	
Material	Carbon steel with internal clad of austenitic stainless steel
Applicable Code	ASME Section III, Class 1

TABLE 5.4-10

Primary Heat Transport Deuteration and De-deuteration System

Design Data

<u>Item</u>	<u>Value</u>
Deuteration and De-deuteration Circuit	
Deuteration and De-deuteration flow rate	0.625 - 1.25 gpm
D ₂ O required to Deuterate 1080 (38 ft ³) resin	56 ft ³
D ₂ O/H ₂ O interface velocity	0.01 to 0.02 ft/min
Applicable Code	ASME Section III, Class 3
Resin Transfer Pump	
Quantity	1
Type	Centrifugal
Duty	Intermittent
Pump Data:	
Fluid	D ₂ O
Flow Rate	75 gpm
Temperature	ambient
Head	210 ft
Design Pressure	150 psig
Design Temperature	150°F
Motor Data:	
Type	Squirrel cage induction motor
Power Supply	Class IV
Construction:	

TABLE 5.4-10 (cont'd)

Primary Heat Transport Deuteration and De-Deuteration System

Design Data

<u>Item</u>	<u>Value</u>
Materials in contact with fluid	Austenitic Stainless Steel
Applicable Code	ASME, Section III, Class 3
D ₂ O and H ₂ O Head Tanks	
Quantity	1 of each
Type	Vertical cylinder, dished ends
Capacity	67 ft ³
Design Conditions:	
Operating Pressure	atmospheric
Operating Temperature	ambient
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Name	Resin Deuteration and De-deuteration Tanks
Quantity	1 of each
Type	Vertical cylinder, dished top, conical bottom
Capacity	52 ft ³

TABLE 5.4-10 (cont'd)

Primary Heat Transport Deuteration and De-deuteration System

Design Data

<u>Item</u>	<u>Value</u>
Design Conditions:	
Operating Pressure	0-150 psig
Operating Temperature	ambient
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 5.4-11

Primary Heat Transport D₂O Sampling System

Design Data

<u>Item</u>	<u>Value</u>
D ₂ O Sampling Circuit	
Types of sample	Hypodermic 300 ml cannister crud
Points sampled	<ul style="list-style-type: none">- reactor inlet and outlet, both circuits- purification inlet to filters outlet of each filter, outlet of each ion exchange column- D₂O collection system
Design Conditions:	
Pressure	1670 psig
Temperature	590°F
Normal Conditions (After cooler):	
Pressure	up to 1630 psig
Temperature	less than 100°F
Applicable Code	ASME Section III, Class 3
Sample Pots:	
Quantity	6
Type	Hypodermic
Material:	
Body	Austenitic Stainless Steel
Septum	Neoprene, Shore hardness 40
Sample Cannisters:	
Quantity	2

TABLE 5.4-11 (cont'd)

Primary Heat Transport D₂O Sampling System

Design Data

<u>Item</u>	<u>Value</u>
Type	Removable pressure vessels
Volume	300 ml
Material	Austenitic Stainless Steel
Crud Collectors:	
Quantity	6
Type	Filtration of suspended solids onto 0.45 micron filter paper
Material	Austenitic Stainless Steel
Pressure reducing coils:	
Quantity	8
Normal Flow	0.31 gpm
Pressure reduced to	150 psig
Sample Coolers:	
Quantity	4
Type	Tube-in-shell
Material	Inconel 600 tubes
Heat exchange area	1 ft ² each

TABLE 5.4-12

PRIMARY HEAT TRANSPORT D₂O COLLECTION SYSTEM DESIGN DATA

<u>Item</u>	<u>Value</u>
Pumps	
Quantity	2
Flow	
Pump Drive HP	
Condenser	
Quantity	1
Volume	
Dimension (L/D)	
D ₂ O Collection Tank	
Quantity	1
Volume	
Dimension (L/D)	

5.4-65

MAIN MODERATOR SYSTEM

Figure 5.4-1

MODERATOR PURIFICATION SYSTEM

Figure 5.4-2

MODERATOR COVER GAS SYSTEM
Figure 5.4-3

MODERATOR D₂O COLLECTION SYSTEM

Figure 5.4-4

5.4-69

MODERATOR DEUTERATION AND DE-DEUTERATION SYSTEM

Figure 5.4-5

5.4-70

MODERATOR D₂O SAMPLING SYSTEM

Figure 5.4-6

Figure 5.4-7 P.H.T.S. PRESSURE AND INVENTORY CONTROL SYSTEM

Figure 5.4-8 PRESSURIZER ARRANGEMENT

Figure 5.4-9 GLAND SEAL CIRCUIT

Figure 5.4-11 P.H.T.S. DEUTERATION & DE-DEUTERATION SYSTEM

Figure 5.4-12 HEAT TRANSPORT D2O SAMPLING SYSTEM

Figure 5.4-13 P.H.T.S. D₂O COLLECTION CIRCUIT

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6.0 ENGINEERED SAFETY FEATURES

6.1 Introduction

Engineered Safety Features (ESF) are designed to minimize the severity of and to mitigate the consequences of postulated accidents assuming failure of the first line of defense against a fission product release.

Safety features are engineered in the plant design to limit radioactive releases to the public for two types of accidents:

- o A single failure in a process system, and
- o A single failure in a process system combined with a coincident failure of one of the safety systems.

The following systems are considered Engineered Safety Features and are independent in design and operation:

Containment Systems

Emergency Core Cooling System (ECCS)

The ECCS is controlled and monitored from the main control room. A limited number of engineered safety features may be controlled from a secondary control center located in the service building, in addition to the main control room.

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Function

The containment structure and associated safety features are designed to contain the energy released during a LOCA, and to dissipate it in such a manner as to preserve the integrity of the structures and the operability of the engineered safety features.

6.2.1.2 Description

The containment structure is a prestressed post-tensioned concrete building comprising three structural components; a base slab approximately 1.55 meters thick; a cylindrical wall approximately 41.5 meters internal diameter with a minimum wall thickness of about 1.07 meters; and a spherical segmented dome with a thickness at the crown of about 0.60 meters.

Beneath the outer dome there is a second dome having an opening in the crown, which together with the perimeter wall forms a container to provide storage for approximately 2,000 cu. meters of water for dousing and emergency core cooling.

To assist in leakage control the containment structure has an internal lining comprised of a plastic coating. All penetrations through the containment envelope are designed to limit leakage to an acceptable level.

The containment structure is structurally separated from the internal structures. This provides flexibility in overall building construction and no interdependence between the containment wall and other structures. The design minimizes the number of attachments to the wall and consequently reduces potential leakage sources.

As shown in Figure 6.2-1, the containment boundary is as follows:

- a) all internal surfaces of the containment building upper dome, cylindrical wall, and base slab
- b) outer surfaces of the spent fuel discharge chamber and its accesses
- c) piping and duct work from the containment inner surface up to and including the second isolation valve
- d) the outer door of containment air locks
- e) closed systems inside containment which are open to the atmosphere outside containment

The containment structure, as well as the above containment boundaries are pressure and leakage tested when erected and tested periodically afterwards.

6.2.1.3 Design Bases

Refer to Table 6.2-1.

6.2.1.4 Safety Evaluation

6.2.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Containment:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 3 - Fire protection.
- GDC 4 - Environmental and missile design bases.
- GDC 16 - Containment design.
- GDC 38 - Containment heat removal.
- GDC 39 - Inspection of containment heat removal system.

- GDC 40 - Testing of containment heat removal system.
- GDC 41 - Containment atmosphere cleanup.
- GDC 42 - Inspection of containment atmosphere cleanup systems.
- GDC 43 - Testing of containment atmosphere cleanup systems.
- GDC 50 - Containment design basis.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 52 - Capability for containment leakage rate testing.
- GDC 53 - Provisions for containment testing and inspection.
- GDC 60 - Control of releases of radioactive materials to the environment.

6.2.1.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with. The program must be expanded to provide assurance that the containment and related systems will function as designed. This will require that the program includes all phases of construction, including the material manufacturer and supplier.

In keeping with the intent of GDC 1, the containment should be designed and constructed to the requirements of ASME Boiler & Pressure Vessel Code, Section III, Division 2. This will place the containment under the jurisdiction of the chief boiler and pressure vessel inspector of the State in which the plant is located.

The seismic capability of the containment, as presently designed, will require a re-evaluation based on the ground accelerations normally assigned to nuclear power plant sites in the USA. In particular, the effect of the

hydrodynamic forces created by the water stored in the dousing tank portion of the containment would have to be evaluated to its effect on the stability of the containment.

The containment would have to be analyzed for the effects of tornadoes, hurricanes, external blast loads, and other phenomena as required of structures sited in the USA.

Sufficient details were not available to fully evaluate the internal compartment structures and ventilating system ducts to assure that hydrogen cannot be trapped in isolated pockets and become explosion/fire hazards. Details were not available as to the types and arrangement of the various fire-protection systems in the containment and other internal compartments. The containment was not assessed for its compliance with GDC 3 as to the types and quantities of flammable materials in the containment used in the construction of the plant.

The use of epoxy-lining on the internal surfaces of the concrete containment is not suitable to prevent damage to the concrete due to the jet force developed during a postulated pipe break of a high-energy piping system. Also, the capability of the concrete structure must be evaluated to withstand unacceptable structural damage or breaching due to internally or externally generated missiles as required by GDC 4.

GDC 16 requires an essentially leaktight barrier against uncontrolled release of radioactivity to the environment. We do not have evaluated lifetime performance of epoxy-linings on concrete surfaces over 30 to 40 years of service. Due to the variable temperature and humidity conditions in the

containment, it would be difficult to positively state that the present lining material would be acceptable. As stated above for GDC 4, epoxy-lining would not be suitable to withstand jet forces from a break in a high-energy pipe system.

The PHWR containment is provided with multiple cooling systems as discussed in 6.2.2. Specific discussions related to the systems meeting GDC 38 are contained in the safety evaluation for each of the systems involved. In addition, discussion of GDC 39, 40, 41, 42 and 43 is discussed in other sections of this report.

Sufficient data is not available to properly assess the design margin available in the containment design to satisfy the requirements of GDC 50. Containment design pressure is determined by rupture of the largest pipe in the primary system, with subsequent blowdown of both coolant loops.

On the basis of the material specifications used for the construction of the metal portions of the containment, GDC 51 requirements for fracture-toughness are not met.

The PHWR containment criteria for leak-rate testing appears to meet the requirements of GDC 52. The containment will fully comply when it is provided with a metal liner in lieu of the epoxy liner now used.

The PHWR containment criteria for testing and inspection will meet GDC 53 after providing a metal liner.

The PHWR containment does not appear to be capable of containing radioactive material as presently designed. The jet force impinging on the epoxy liner could cause a breach in containment. On this basis, GDC 60 is not fully complied with in the present design.

6.2.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.

The containment design and construction should comply with the requirements of ASME Boiler & Pressure Vessel Code, Section III, Division 2.

GDC 2 - No change recommended pending review of structural analysis.

GDC 3 - Insufficient data available to accept or recommend design changes.

GDC 4 - Provide metal liner in lieu of epoxy liner to mitigate effect of jet force due to pipe break and internally generated missiles.

GDC 16 - Provide metal liner to provide a leaktight containment under postulated accident conditions.

GDC 38 - See Section 6.2.2.

GDC 39 - See Section 6.2.2.

GDC 40 - See Section 6.2.2.

GDC 41 - See Section 6.2.2.

GDC 42 - See Section 6.2.2.

- GDC 43 - See Section 6.2.2.
- GDC 50 - Provide a reasonable margin over the maximum accident conditions for setting the design conditions.
- GDC 51 - Provide material for the containment liner, access openings, and penetrations that meet the fracture-toughness requirements of this criterion and Section III of the ASME B&PV Code.
- GDC 52 - No change if metal liner is provided.
- GDC 53 - No change if metal liner is provided.
- GDC 60 - Provide a metal liner.

6.2.2 Containment Heat Removal System

The containment heat removal system is comprised of both passive and active systems that are designed to preserve the integrity of the containment boundary by removing thermal energy from the containment in the event of a Loss of Coolant Accident (LOCA) or a Steam Line Break (SLB).

The specific systems involved are the ECCS, discussed in Section 6.3, the Containment Dousing System, the Containment Fan Cooler System, and the Shut-down Cooling System, which are discussed below.

6.2.2.1 Containment Dousing System

6.2.2.1.1 Function

The Containment Dousing System (CDS) is designed to remove the energy released to the containment in the event of a LOCA or SLB in order to prevent the containment pressure from exceeding the design pressure.

6.2.2.1.2 Description

A schematic of the Containment Dousing System is shown in Figure 6.2-2. The entire Dousing System is located inside the containment.

Dousing water temperature and level are monitored during operation. A system of cooling pipes in the dousing tank is provided to maintain the tank water temperature at 85 F or less. Valves and piping for the system are designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code for Class 2 components, and are designed to withstand the design basis earthquake.

The Dousing System is initiated automatically by a high pressure signal (9 psig) from 2 out of 3 sensors. Approximately 540,400 gallons is provided for containment dousing, and 132,600 for the Emergency Core Cooling System from the dousing tank. The dousing valves close at an overpressure of 3 psig. The dousing system can be manually operated from the control room as well.

The active components of the system are comprised of two independent and testable sets of valves, each set being of a different manufacture. Each set of six valves is arranged with two valves in series on three headers. There are six headers and twelve valves.

Embedments in the tank for dousing and Emergency Core Cooling supply headers are designed to transmit forces and moments from piping systems due to thermal and earthquake events into the tank. These embedments have flanges at each end to mate with the epoxy liner inside the tank and on its lower surface.

6.2.2.1.3 Design Bases

Refer to Table 6.2-2.

6.2.2.1.4 Safety Evaluation

6.2.2.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Containment Dousing System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.

GDC 38 - Containment heat removal.

GDC 39 - Inspection of containment heat removal system.

GDC 40 - Testing of containment heat removal system.

6.2.2.1.4.2 Assessment

The Quality Assurance Program and record keeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with. Additional comments are in-common with those in Section 6.2.1.4.2.

The seismic capability of the dousing tank and piping systems associated with the dousing system should be evaluated for compliance with GDC 2. A steel liner should be incorporated in the dousing tank design.

It should be noted that Canadian design criteria does not superimpose an SSE on a postulated LOCA as is the case in the U.S. Therefore, the piping system associated with the downcomers of the dousing system is not seismically qualified.

Protection against the effects of internally generated missiles for the dousing system has not been provided. This can be done through the use of barriers or in providing redundancy so that the system function will not be impaired to meet the requirements of GDC 4.

In general, the dousing system meets the requirements of GDC 38. The use of relatively large orifices for spray distribution appears to provide an inefficient method of removing heat from the containment atmosphere.

There is no chemical addition to the dousing water to provide for iodine removal.

Provisions have not been incorporated for inservice inspection of the dousing tank and associated piping systems as required by GDC 39 and GDC 40 and ASME Section XI, as applicable.

6.2.2.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Provide seismic restraints for the dousing system piping and a steel liner for the tank.
- GDC 4 - Evaluate the PHWR design to assure that the system is provided with missile protection.
- GDC 38 - Evaluate heat removal provided with present spray nozzle design.
- GDC 39 - Provide inservice inspection program for dousing tank and associated piping system.
- GDC 40 - See GDC 39, above.

6.2.2.2 Containment Fan Cooler System

6.2.2.2.1 Function

The Fan Cooler System will reduce containment pressure and temperature to ambient, subsequent to the operation of the ECCS following a LOCA or SLB.

6.2.2.2.2 Description

This system consists of fan coolers on Emergency Class III power and having Class III process water supply to the coils.

The fan motors are designed to handle the steam/air mixture at containment design temperature and pressure.

6.2.2.2.3 Design Bases

Information is not available.

6.2.2.2.4 Safety Evaluation

6.2.2.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Containment Fan Cooler System:

- o GDC 38 - Containment Heat Removal.

6.2.2.2.4.2 Assessment

Fan cooler systems have been analytically demonstrated to be an effective means for energy removal following an accident but not in limiting the initial pressure peak. The design of this system is almost identical to the fan cooler systems employed in LWR containments.

In particular, for certain emergency conditions such as loss-of-coolant-accident, and a passive failure of the Recirculated Water System piping,

all cooling water to the containment fan coolers. In our assessment, the cooling water system design provides a mechanism for failure propagation of containment heat removal systems which may lead to gross failure of systems designed to prevent design parameters from being exceeded. Refer to discussions on cooling water systems in Chapter 9.0.

6.2.2.2.4.3 Recommended Design Change

The following design changes are recommended:

- o Eliminate Emergency Fan Cooling System as a safety-related system. Qualify components of the system as Seismic Category I if in the containment.

6.2.2.3 Shutdown Cooling System

6.2.2.3.1 Function

The Shutdown Cooling System is designed to remove decay heat from the reactor core by circulating and cooling the Primary Heat Transport System. It is also designed to provide core cooling with the Primary Heat Transport System drained to the headers to permit maintenance of the steam generators and Heat Transport Pump internals.

6.2.2.3.2 Description

As shown in Figure 6.2-3, the Shutdown Cooling System consists of two 100% capacity pumps and two 100% capacity heat exchangers. One pump and one heat exchanger are located at each end of the calandria, connected between the inlet and outlet headers of both heat transport circuits. The system is normally full of D_2O and is normally isolated from the Primary Heat Transport System by 8 sets of duplicated valves. Emergency Core Cooling is injected into the inlet headers via the Shutdown Cooling System piping.

During a normal cooldown operation, steam from the steam generators flows through the turbine bypass valves into the condensers for approximately 30 minutes. Cooling water from the Raw Service Water System removes the heat energy to the ultimate heat sink during this mode of operation.

Cooldown to cold shutdown is achieved by opening the isolation valves at the headers and directing a portion of the flow through the Shutdown Cooling System heat exchangers. During the initial phase, the Shutdown Cooling pumps are not in operation, but bypassed and water is circulated through the Shutdown Cooling System heat exchangers by using the Primary Heat Transport System pumps. In the later stages of cooldown, the shutdown cooling pumps

are used since the heat addition from the primary pumps becomes the dominant heat source.

When both shutdown cooling pumps are operating and the Primary Heat Transport System is full, a part of the shutdown cooling flow will bypass the core through the steam generators. The shutdown cooling pumps are sized such that no boiling occurs in any of the pressure tube channels. When some heavy water is removed and the steam generators are partially emptied, all of the flow passes through the core.

If the Primary Heat Transport System is open for maintenance of the pumps or steam generators when one Shutdown Cooling pump becomes unavailable, the other pump is tripped, due to the fact that one pump cannot provide sufficient flow to keep the water from rising above the headers into the Primary Heat Transport loops. To prevent reverse flow through the Shutdown Cooling pumps when the Primary Heat Transport System pumps are operating, a bypass line is connected between each Shutdown Cooling Pump suction and discharge. During such operation, the gate valves in the bypass lines are opened and the Shutdown Cooling Pump isolation valves are closed. During operation with one Shutdown Cooling Pump, a bypass line is provided between the pump suction and the heat exchanger outlet.

The Shutdown Cooling pumps are vertical centrifugal, 350 HP pumps which are provided with Class III electrical power.

Temperature controllers are provided to hold the Primary Heat Transport System temperature at a constant value for an indefinite period by varying the flow of the Recirculated Water System. The temperature control on each Shutdown Cooler consists of a single channel analog loop with panel mounted

controller. These loops control the temperature of the D₂O at the outlet of the coolers when the Shutdown Cooling pumps are operating. When the Heat Transport pumps are operating on cooldown, the flow through the coolers will be reversed and the temperature will be sufficiently above the set point of the temperature controller so that the Recirculated Service Water control value will be held fully open.

The heat transport system is drained by utilizing the shutdown cooling pumps; flow is taken from downstream of the shutdown cooling pumps and sent via the purification system to the D₂O storage tank. Feed and bleed is used to control the D₂O level in the heat transport system after the level is reduced. The feed pumps are used to refill the heat transport system.

6.2.2.3.3 Design Bases

Refer to Table 6.2-3.

6.2.2.3.4 Safety Evaluation

6.2.2.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Shutdown Cooling System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environment and missile design bases.
- GDC 29 - Protection against anticipated operational occurrences.
- GDC 34 - Residual heat removal.
- GDC 35 - Emergency core cooling.
- GDC 36 - Inspection of emergency core cooling system.

GDC 37 - Testing of emergency core cooling system.

6.2.2.3.4.2 Assessment

The Quality Assurance Program and record keeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with.

The components of the system should be evaluated as to their seismic capability, per GDC 2, for the ground accelerations normally assigned to nuclear power plant sites in the USA.

Separation of redundant systems and protection against the effect of internally generated missiles should be evaluated for compliance with GDC 4.

This system meets the intent of GDC 29 in that it possesses redundancy in components and capability in equipment to perform its function, however, piping arrangement would have to be revised to comply fully.

GDC 34 and 35 are applicable since ECC is injected through the Shutdown Cooling System piping and also that the system is used for residual heat removal. Those portions of the criteria that are applicable to the Shutdown Cooling System are complied with.

An inservice inspection program should be provided for the inspection of system components and functional testing of pumps and valves. This is required by GDC 36 and 37 and by 10 CFR 50.55a.

6.2.2.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - No change recommended pending review of seismic analysis.
- GDC 4 - Insufficient data available to accept or recommend design changes.
- GDC 29 - Revise piping to meet single failure criteria.
- GDC 34 - No change.
- GDC 35 - No change.
- GDC 36 - Provide an inservice inspection program.
- GDC 37 - See GDC 36, above.

6.2.3 Containment Isolation System

6.2.3.1 Function

In the event of a LOCA or SLB, there exists the possibility for a radiological release to the peripheral buildings and in turn to the environment through the break during containment pressurization. To prevent such an occurrence, containment isolation valves must be provided in accordance with General Design Criteria 54 through 57 of Appendix A of 10 CFR 50.

6.2.3.2 Description

Some systems are equipped with isolation valves, such as the Moderator Purification System, although information regarding the operating modes and methods of operation is not available.

6.2.3.3 Design Bases

Information is not available.

6.2.3.4 Safety Evaluation

6.2.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Containment Isolation System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 54 - Piping systems penetrating containment.
- GDC 56 - Primary containment isolation.
- GDC 57 - Closed system isolation valves.

6.2.3.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with. Piping penetrating containment and its isolation valves shall be classified as ASME Class 1 or Class 2, depending on the classification of the process system.

The seismic capability of the piping and valves forming the containment isolation system should be evaluated for compliance with GDC 2.

The physical design of the piping and valves forming the containment isolation should be evaluated for compliance with GDC 4: missile protection.

The material for the piping and valves forming the containment isolation should be selected on the basis of its fracture-toughness for compliance with GDC 51.

The piping penetrating containment must be provided with leak detection. An inservice inspection program should be provided to periodically inspect the pipe and test the operability of the isolation valves in accordance with GDC 54.

Piping that connects to the containment atmosphere shall be provided with isolation valves inside and outside the containment in accordance with GDC 56.

Each line of a closed system that penetrates the containment shall be provided with an isolation valve outside of containment in accordance with GDC 57.

6.2.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria.

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - No change recommended pending review of seismic analysis.
- GDC 4 - Insufficient data available to accept or recommend design changes.
- GDC 51 - Select material for the piping that penetrates containment and its isolation valves for its fracture-toughness properties to meet this criterion.
- GDC 54 - Provide leak detection of all piping penetration, an inservice inspection program for the piping, and periodic operation of the isolation valves to test function.
- GDC 56 - Provide automatic or remote manually operated valves, both inside and outside of containment, for piping connected to the containment atmosphere.
- GDC 57 - Provide an automatic or remote manually operated valve on the outside of containment for piping of closed systems.

6.2.4 Combustible Gas Control System

6.2.4.1 Function

The purpose of this system is to prevent the accumulation of hydrogen gas from reaching combustible concentrations in the Reactor Building following a loss of coolant accident. Hydrogen may accumulate in the Reactor Building as a result of:

- 1) Metal-water reaction between zirconium and reactor coolant.
- 2) Radiolytic decomposition of the post accident emergency coolant and dousing water.

6.2.4.2 Description

A combustible gas control system is not provided.

6.2.4.3 Design Bases

Information is not available.

6.2.4.4 Safety Evaluation

6.2.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Combustible Gas Control System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 41 - Containment atmosphere cleanup.
- GDC 42 - Inspection of containment atmosphere cleanup systems.
- GDC 43 - Testing of containment atmosphere cleanup systems.
- GDC 50 - Containment design basis.

GDC 51 - Fracture prevention of containment pressure boundary.

GDC 53 - Provisions for containment testing and inspection.

GDC 54 - Piping systems penetrating containment.

GDC 56 - Primary containment isolation.

6.2.4.4.2 Assessment

The PHWR is not provided with a combustible gas control system.

6.2.4.4.3 Recommended Design Change

The following design change is recommended in order to have the PHWR comply with the applicable NRC criteria listed in Section 6.2.4.4.3, above:

- o Provide a Combustible Gas Control System and implement the applicable NRC General Design Criteria.

6.2.5 Containment Leakage Testing

6.2.5.1 Function

Provisions for leakage testing of the primary containment and systems penetrating primary containment are required to assure the availability of the containment function, when required.

6.2.5.2 Description

Periodically, leak rate tests of the containment and its penetrations, will be made to assure containment function.

6.2.5.3 Design Bases

Overall leakage of the containment system will not exceed 0.5% of the building volume per day at 18 psig.

6.2.5.4 Safety Evaluation

6.2.5.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Containment Leakage Testing:

GDC 1 - Quality standards and records.

GDC 16 - Containment design.

GDC 52 - Capability for containment leakage rate testing.

GDC 53 - Provisions for containment testing and inspection.

6.2.5.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are required for the periodic testing of the containment.

To assure having an acceptable leak rate capability of the containment, provisions must be incorporated to perform the necessary tests. This is required to conform with GDC 16, 52, 53, and with Appendix J of 10 CFR 50.

6.2.5.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria.

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 16 - Provide a plan and procedure for conducting leak tests of all penetrations and a leak rate test of the containment on a periodic basis.
- GDC 52 - See GDC 16, above.
- GDC 53 - See GDC 16, above.

TABLE 6.2-1

Containment Design Data

<u>Item</u>	<u>Data</u>
Type	Prestressed, post-tensioned reinforced cylindrical concrete
Inside Diameter	136 ft
Inside Height	168 ft
Design Pressure	18 psig
Liner	epoxy
Wall Thickness	3.5 ft
Dome thickness (minimum)	2.0 ft
Design leak rate	0.5 vol. %/day

TABLE 6.2-2

Dousing System Design Data

<u>Item</u>	<u>Data</u>
General:	
Water Source	Demineralized Light Water
Water Temperature	80°F
Actuation Pressure	
Initial Peak Dousing Flow Rate	108,000 gpm
Dousing Tank:	
Quantity	1
Form	Prestressed Concrete Integral with Reactor Building Dome
Diameter	
Capacity	672,000 gallons

TABLE 6.2-3Shutdown Cooling System Design Data

<u>Item</u>	<u>Data</u>
Heat Exchangers:	
Quantity	2
Type	U-tube in shell
Duty	Intermittent
Capacity	50%
Heat Transferred	42×10^6 Btu/lb
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	13,280 gpm
Inlet Temperature	130°F
Outlet Temperature	104°F
Design Pressure	1870 psig
Design Temperature	535°F
Shell Side Data	
Fluid	H ₂ O
Flow Rate	4,800 gpm
Inlet Temperature	77°F
Outlet Temperature	94°F
Design Pressure	150 psig
Design Temperature	200°F
Construction	
Tube Material	Incaloy 800
Tube Outside Diameter	.625 inches
Shell Material	Carbon Steel

TABLE 6.2-3 (cont'd)

Shutdown Cooling System Design Data

<u>Item</u>	<u>Data</u>
Applicable Codes	
Tube side	ASME Section III, Class 1
Shell side	ASME Section VIII
Pumps and Motors	
Quantity	2
Type	Vertical, Centrifugal
Duty	Intermittent
Capacity	50%
Pump Data:	
Fluid	D ₂ O
Flow Rate	3,000 gpm
Temperature	130°F
Head	200 ft
Design Pressure	1870 psig
Design Temperature	535°F
Motor Data:	
Type	Vertical Squirrel Cage Induction
Power Supply	Class III, 250 HP
Construction:	
Materials in contact with fluid	Carbon Steel
Applicable Code	ASME Section III, Class 1

CONTAINMENT SYSTEM

Figure 6.2-1

DOUSING AND EMERGENCY COOLING SYSTEM
Figure 6.2-2

SHUTDOWN COOLING WATER SYSTEM
FIGURE 6.2-3

6.3 Emergency Core Cooling System (ECCS)

6.3.1 Function

The purpose of this system is to provide cooling water to the reactor core to compensate for loss of normal capacity following a loss-of-coolant-accident (LOCA).

6.3.2 Description

A schematic diagram of the ECCS is shown in Figure 6.3-1. In the event of a LOCA, the pressure increase inside the reactor building initiates a signal which closes the valves in all the interconnection lines joining the two heat transport loops, thereby isolating the unfailed loop from the failed loop. In addition, the heat transport main coolant pumps would be stopped and the steam discharge valves opened. Such actions initiate an emergency cooldown and pump coastdown before emergency core cooling begins. When the pressure in the failed heat transport loop drops below 100 psig, the emergency light-water injection valves open and annunciation is given in the control room. Check valves prevent outflow of D_2O . Emergency injection flow will begin when the pressure in the ruptured loop is less than the pressure due to the static head of water in the Dousing Tank (approximately 38 psig).

For emergency core cooling, as shown in Figure 6.3-1, two lines leading from the Dousing Tank are equipped with normally closed injection valves. Any leakage of light-water past the valves is collected from the downstream piping and directed to the Reactor Building light-water collection system. Test valves, downstream of the injection valves, are normally open.

Downstream of the test valves, the lines branch to each end of the reactor through a check valve. At each end of the reactor, a further sub-branch

supplies each pair of inlet and outlet headers, with normally closed isolation valves being used to interconnect the ECCS to the heat transport system. These valves are duplicated for reliability.

Upon depletion of the stored water in the Dousing Tank, long term core cooling is continued by pumping water from the Reactor Building sump back into the core. Building sump level is monitored and switchover is done manually when sufficient NPSH exists. Two 100% pumps and one 100% heat exchanger is provided. The pumps and heat exchanger are external to the Reactor Building, with isolation valves installed on pump suction and pump discharge piping for each pump. The discharge of the ECCS recovery piping is upstream of the injection valves mentioned above so that any excess water can be returned to the Dousing Tank.

Cooling water to the heat exchanger is provided by service water or, in event of loss of normal service water due to a seismic event, by the Emergency Water Supply System, described in Section 9.2.2. Common piping of the Service Water and EWSS is seismically qualified but not redundant.

The light-water emergency injection valves can be tested while the reactor is operating by first closing the downstream isolating valve and then opening and closing each of the light water emergency injection valves.

The D₂O emergency cooling valves can be tested while the reactor is operating by first closing the isolating valve at the check valve at one end of the reactor and then opening and closing each D₂O emergency cooling valve at the same end of the reactor.

All the equipment up to the check valves exclusive, is designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code for Class

2 components. The remainder of the system is designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code for Class 1 components. In addition, all components are seismically qualified.

During the above sequence of events, the unruptured loop is cooled by the steam generators with feedwater being supplied as discussed in Section 10.2. Feedwater to the unfailed loop is made available by remote manually opening the isolation valves in the feedwater lines, which close automatically on high Reactor Building pressure.

6.3.3 Design Bases

Refer to Table 6.3-1.

6.3.4 Safety Evaluation

6.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Emergency Core Cooling System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 35 - Emergency core cooling.
- GDC 36 - Inspection of emergency core cooling system.
- GDC 37 - Testing of emergency core cooling system.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 54 - Piping systems penetrating containment.
- GDC 55 - Reactor coolant pressure boundary penetrating containment.
- GDC 57 - Closed system isolation valves.

6.3.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with.

The seismic capability of the ECC System, as presently designed, will require a re-evaluation based on the ground accelerations normally assigned to nuclear power plants sited in the USA to comply with GDC 2.

Protection against the effects of missiles generated both inside and outside the containment should be evaluated to assure that the components and the system can function properly when required. Since this is not a high-energy system, pipewhip restraints will not be required, in accordance with GDC 4.

This system, as presently designed, does not comply with GDC 35. The ECC System has only one heat-exchanger and a common recirculating water piping system. The system does not provide redundancy therefore, it does not provide assurance that it can meet the single-failure criterion, as required in the USA.

An inservice inspection program is required to assure that the system is capable of performing its required function without failure, when required. The ISI program should comply with the requirements of GDC 36 and 10 CFR 50.55a.

The ISI program should include provisions for periodically testing the function of the pumps and valves of the system to assure that they will be operable when required. The program should be in accordance with GDC 37 and 10 CFR 50.55a.

For a discussion of the piping systems that penetrate the containment, see Section 6.2.3.4.2 that assesses the present design of the Containment Isolation System. That portion of the piping system penetrating containment is treated as having a second function, that of being part of the containment isolation systems, and therefore must comply with GDC 51, 54, 55, and 57.

6.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.

- GDC 2 - No change recommended pending review of seismic analysis.
- GDC 4 - Insufficient data available to accept or recommend design changes.
- GDC 35 - Provide a redundant 100-percent heat exchanger, cooling water supply, recirculation piping and valves to have the system comply with the single-failure criteria. Isolation of the cold side of the heat exchanger is required since the system may contain fission products, (see Section 9.2.2 for additional details).

The single check valve used to prevent flow from one loop to the other loop of the reactor coolant system should have an additional automatic-operated valve in line to assure isolating the coolant loops.

- GDC 36 - Provide an inservice inspection program that provides for accessibility to components, inspection of welds of the pressure boundaries, and functional testing of pumps and valves in the system.
- GDC 37 - See GDC 36, above.
- GDC 51 - See GDC 51, Section 6.2.3.4.3.
- GDC 54 - See GDC 54, Section 6.2.3.4.3.
- GDC 55 - See GDC 55, Section 6.2.3.4.3.
- GDC 57 - See GDC 57, Section 6.2.3.4.3.

TABLE 6.3-1

Emergency Core Cooling System Design Data

<u>Item</u>	<u>Data</u>
Heat Exchangers:	
Quantity	2
Type	Horizontal U-tube in shell
Duty	Intermittent
Capacity	100%
Heat Transferred	143.0×10^6 Btu/hr
Tube Side Data	
Fluid	H ₂ O and D ₂ O Mixture
Flow Rate	9600 gpm
Inlet Temperature	150°F
Outlet Temperature	120°F
Design Pressure	150 psig
Design Temperature	220°F
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	12,000 gpm
Inlet Temperature	65°F
Outlet Temperature	89°F
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Tube Material	Carbon Steel
Shell Material	

TABLE 6.3-1 (cont'd)

Emergency Core Cooling System Design Data

<u>Item</u>	<u>Data</u>
Applicable Codes:	
Tube Side	ASME Section III, Class 2
Shell Side	ASME Section VIII
Pumps and Motors:	
Quantity	2
Type	Vertical, Multi-Stage Sump Type
Duty	Intermittent
Capacity	100%
Pump Data:	
Fluid	H ₂ O and D ₂ O Mixture
Flow Rate	9600 gpm
Temperature	150°F
Head	200 ft
Design Pressure	150 psig
Design Temperature	150°F
Motor Data:	
Type	3 Phase, Squirrel Cage Induction
Power Supply	Class III
Construction	
Materials in contact with fluid	Carbon Steel
Applicable Code	ASME Section III, Class 2

EMERGENCY CORE COOLING

Figure 6.3-1

6.4 Control Room Habitability System

6.4.1 Function

The purpose of the Control Room Habitability System is to assure accessibility and habitability of the control room subsequent to the occurrence of postulated accidents resulting in release of fission products and/or hazardous chemical releases such as chlorine.

6.4.2 Description

Two 100% capacity air conditioning units are provided, each with its own air filter. Proportioning dampers, controlled by temperature sensors in the fresh air duct automatically reduce the outside air rates when the outside air exceeds 70 F. Relative humidity is maintained at $50\% \pm 10\%$.

In the event of fission product ingress, the control room is isolated and the operating personnel are provided with emergency breathing apparatus.

6.4.3 Design Bases

Information is not available.

6.4.4 Safety Evaluation

6.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Control Room Habitability System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 19 - Control room.

6.4.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with.

The seismic capability of the control room, equipment contained therein, and the habitability system equipment should be evaluated to assure that they are acceptable based on ground accelerations normally assigned to nuclear power plants sited in the USA.

The vulnerability of the control room to internal or external missiles should be evaluated to assess their effect on the habitability system to comply with GDC 4.

The present design of the control room does not comply with GDC 19 in that a habitability system is not provided as required for nuclear power plants sited in the USA. There are no remote fresh air intakes provided, nor is the control room provided with toxic chemical detectors that could cause sealing-off the fresh air intakes. The present design incorporates the use of PVC in the cable spreading room that, in itself, could be a source of local toxic fumes should there be a fire. The air-conditioning system appears to meet the requirements of this service but fission product particle filters have not been provided to assure habitability if radiation were released from the containment.

Although GDC 19 does not address this particular issue, the present design should be evaluated to deny entrance of unauthorized personnel to the control room.

6.4.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - No change recommended pending review of the seismic analysis and seismic qualification of equipment.
- GDC 4 - Insufficient data available to accept or recommend design changes.
- GDC 19 - Redesign control room to include:
 - o Remote air intakes with provisions for missile protection.
 - o Provide filters on air conditioning equipment to remove radioactive particulates.
 - o Install toxic chemical detectors and alarms and provisions to seal-off control room.
 - o Evaluate the possibility of generating toxic fumes in the control room as a result of burning PVC in the cable spreading area.

6.5 Fission Product Removal System

6.5.1 Function

The purpose of the fission product removal system is to remove fission products from building atmosphere following an accident. Such areas normally include containment, areas where the gas treatment system is located, the emergency air cleaning systems for the fuel handling area, and the control room.

6.5.2 Description

The cleanup system provided consists of a network of exhaust ducting inside the Reactor Building with a single penetration through the containment boundary to the second floor of the Service Building. At each penetration of the containment boundary, both the supply and return air ductwork is equipped with a double set of isolation dampers. The dampers are designed to be capable of closing against containment design pressure. Control room operation and status of the dampers is provided.

A bank of activated carbon filters, preceded by absolute filters are provided in the Service Building to permit continuous or intermittent cleanup of the exhaust air. Fans are located at the clean air side of the filter bank.

6.5.3 Design Bases

Information is not available.

6.5.4 Safety Evaluation

6.5.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Fission Product Removal System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 16 - Containment design.
- GDC 41 - Containment atmosphere cleanup.
- GDC 42 - Inspection of containment atmosphere cleanup systems.
- GDC 43 - Testing of containment atmosphere cleanup systems.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 52 - Capability for containment leak rate testing.
- GDC 54 - Piping systems penetrating containment.
- GDC 56 - Primary containment isolation.
- GDC 57 - Closed system isolation valves.
- GDC 60 - Control of releases of radioactive materials to the environment.

6.5.4.2 Assessment

Insufficient information available to perform a safety evaluation.

6.5.4.3 Recommended Design Change

Insufficient information available to recommend acceptability or design changes.

6.6 Inservice Inspection

6.6.1 Function

Inservice Inspection includes the procedures, examinations and evaluations, whereby safe and continued nuclear power plant operation can be assured.

The components must be examined periodically and indications evaluated in a conservative manner. Defects can be corrected to assure that the plant's quality is maintained to the same level to which it was constructed.

6.6.2 Description

The Inservice Inspection Program is not described in sufficient detail for the plant evaluated in this report.

6.6.3 Design Bases

Insufficient data available to determine design bases used for the HWR plant evaluated in this report.

6.6.4 Safety Evaluation

6.6.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the Inservice Inspection Program:

- GDC 1 - Quality standards and records.
- GDC 32 - Inspection of reactor coolant pressure boundary.
- GDC 36 - Inspection of emergency core cooling system.
- GDC 37 - Testing of emergency core cooling system.
- GDC 39 - Inspection of containment heat removal system.
- GDC 40 - Testing of containment heat removal system.
- GDC 42 - Inspection of containment atmosphere cleanup systems.

GDC 43 - Testing of containment atmosphere cleanup systems.

GDC 45 - Inspection of cooling water system.

GDC 46 - Testing of cooling water system.

GDC 53 - Provisions for containment testing and inspection.

GDC 54 - Piping systems penetrating containment.

6.6.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with.

The present PHWR plants do not require an inservice inspection program as required by USA regulations. In order to comply with the laws, codes, regulations, etc. of the USA, it will be necessary to compile laws now, before additional PHWR plants are constructed.

The program should be structured as to permit its implementation during the conceptual design of the plant. Access to, and around, essential component should be provided when initially laying-out the components in the plant. During the conceptual plant design phase, consideration should be given as to what type of nondestructive examination will be given and to how the testing will be done (hands-on or remotely). Also during the planning, consideration should be given to the radiation-level at and around the components to be examined and if portable shielding will be required and how it can be erected in place.

Consideration should also be given to which pumps, valves, and other components must be operated to test their operability.

The inservice inspection program is required by 10 CFR 50.55a, the various GDC's listed in Section 6.6.4.1, above, and by the States that have accepted, by law, ASME Section XI, of the Boiler & Pressure Vessel Code.

6.6.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.

GDC's listed in Section 6.6.4.1, above.

Review existing plant design to ascertain if equipment subject to inservice inspection is accessible and also determine the radiation level at and around the component.

Prepare a complete inservice inspection program that details what is to be inspected, how it is to be inspected, extent and frequency of examination, evaluation standards of examinations, procedures for inservice inspection, and other steps as developed.

Consideration should also be given to the special machinery and fixtures that must be designed and used in implementing the inservice inspection.

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CHAPTER 7.0

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Appendix 7-A

7.0 INSTRUMENTATION AND CONTROLS

7.1 Identification of Safety Related Systems

7.1.1 Function

Safety-related systems which directly relate to public safety are:

a. Protection Systems

1. Reactor Trip System

This system initiates a reactor trip, if unsafe conditions should develop.

2. Engineered Safety Features Actuation System (ESFAS)

This system is required to function under accident conditions to mitigate the consequences of an accident.

b. Systems required for Safe Shutdown

These systems are required for safe shutdown under emergency conditions (SSE, Control Room Fire, etc.). These systems contain the automatic and manual devices required to reach and maintain a hot shutdown with provisions to go subsequently to a cold shutdown.

c. Safety Related Display Instrumentation

Safety-Related Display Instrumentation comprises those indicators that provide the operator with vital information to aid in the safe shutdown of the plant, to maintain the plant in a safe hot shutdown condition and to remove decay heat after postulated incidents, such as LOCA.

d. All Other Systems Required for Safety

These are all other instrumentation systems required for safety but have not been mentioned previously under points a through c..

These include:

- Decay heat suction valve interlock
- Core flooding isolation valve interlock
- Decay heat cooler outlet and bypass flow control
- Steam line break instrumentation and control
- Containment atmosphere monitoring
- Refueling interlocks
- Startup control interlocks

7.1.2 Description

Sections 7.2 through 7.6 describe all previously mentioned safety-related systems.

7.1.3 Design Bases

Applicable parts of Sections 7.2 through 7.6 describe the Margins in Design, Applicable Codes and Standards used for designing various safety-related systems.

7.1.4 Safety Evaluation

7.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Instrumentation and Controls for Safety-Related Systems:

See Section 7.1 of Appendix 7-A of this report for a listing of the applicable General Design Criteria.

7.1.4.2 Assessment

None.

7.1.4.3 Recommended Design Change

This section does not address a particular system.

7.2 Reactor Trip System

7.2.1 Function

The Reactor Trip System initiates a reactor trip when a sensed parameter (or a group of parameters) exceeds a setpoint value indicating the approach of an unsafe condition. In this manner, the reactor and reactor coolant system is protected from overpressurization and fuel cladding damage.

7.2.2 Description

The scope of the reactor trip system includes all electronics, signal processing equipment and cabling, from the system sensor to the input terminals of the reactor trip system. The Heavy Water Reactor has two reactor trip systems. They are designated as Shutdown System No. 1 and No. 2. Figure 7.2-1 illustrates the shutdown system parameters and their relation to the rest of the plant.

7.2.2.1 Shutdown System No. 1 (SDS 1)

The primary method of quickly terminating reactor operation when certain parameters enter an unacceptable range is the release of 28 spring-assisted gravity-drop shutoff rods. Shutdown system number one (SDS 1) employs an independent triplicated logic system, which senses the requirement for reactor trip and de-energizes the direct current clutches to release the shutoff rods.

Trip parameters for SDS 1 are identified in Table 7.2-1. The system has sufficient reactivity depth and acts with sufficient speed that the Canadian reactor siting criterion (maximum Iodine-131 release to the public for all single failures) is met. Twenty-six of the 28 shutoff rods introduce approximately 55 milli-k of negative reactivity in two seconds.

Table 7.2-1 shows that there are four measured parameters which will initiate a reactor trip through SDS 1. The selection of parameters is such that there are redundant sensing parameters for eight of the nine categories of process failures identified. For the other category, i.e. small loss-of-coolant, for which there is sufficient time for operator intervention, initiation of a trip manually by the operator provides backup. An alarm window is provided in this case to indicate to the operator the requirement for manual trip. A reactor stepback is also provided which will rapidly reduce reactor power prior to any requirement for shutdown system action (for further details on Reactor Stepback, see Section 7.7.2.3).

Each parameter is measured at three separate locations, each measurement going to one of the three independent trip channels.

7.2.2.2 Shutdown System No. 2 (SDS 2)

A second method of quickly terminating reactor operation when certain combinations of very unlikely failures occur is the rapid injection of concentrated gadolinium nitrate solution into the bulk moderator through six horizontally distributed nozzles. This second shutdown system (SDS 2) employs an independent triplicated logic system which senses the requirement for this emergency shutdown and opens fast-acting helium pressure valves to inject the gadolinium poison into the moderator.

Trip parameters for SDS 2 are identified in Table 7.2-2(b).

There are two measured parameters which will initiate a reactor shutdown through SDS 2. The parameters are high neutron power derived from horizontal in-core detectors and high neutron log rate derived from horizontal ion chambers. Both these parameters will be effective in sensing the accident conditions identified in Table 7.2-2.

7.2.2.3 Initiating Systems and Circuits for SDS 1

7.2.2.3.1 Ion Chambers

The arrangement of the three horizontal in-core flux detectors is shown in Figure 4.6.10.

Three uncompensated ion chambers, located in separate housings on the east side of the calandria are provided for SDS 1. Each housing also contains one regulating system ion chamber, but in a separate cavity within the housing, and a test shutter. Care has been taken to minimize the possibility of ingress of light water into the ion chamber cavities.

The test shutter is the conventional piston-operated boral sleeve and will have the capability of increasing the flux to the ion chamber by up to 40 percent. The piston speed is adjustable to provide the necessary neutron rate signals. A shutter test initiated from the control room checks one SDS 1 and one regulating system ion chamber simultaneously.

The output current from each ion chamber goes to an amplifier which will produce log neutron power, linear neutron power and rate log signals of which only the last one goes to the shutdown system.

7.2.2.3.2 Vertical In-Core Detectors

The arrangement of the vertical in-core flux monitoring assemblies is shown in Figure 4.6-10.

The assemblies have single-ended platinum detectors wound on them to provide high neutron power signals to SDS 1. These include provision for spares. The detectors have been located so that they are not excessively

close to adjusters, control absorbers and zone control units to minimize any interaction effects. The detectors are coiled onto the assemblies over about two lattice pitches and are separate from any regulating system detectors; the detectors for each channel of SDS 1 are on different assemblies. The assemblies have also been located so that potential incidents inside the calandria, such as a failed pressure tube, would not disable the signal to more than one shutdown channel.

The current output of the detector goes to a linear amplifier producing an output covering the range 5% to 150% full power.

Test facilities are provided in the control room to check the trip circuit by injection of a test current on the amplifier input. The detector outputs are displayed in the control room for purpose of monitoring the signals for correctness at power and during power maneuvering. There is also a facility for checking the insulation resistance of each detector. This resistance is a measure of detector integrity.

7.2.2.3.3 Heat Transport Pressure

Heat transport pressure is measured at three widely separated locations on each of the four reactor outlet headers for a total of 12 measurements. The pressure transmitters are mounted on the channelized instrument racks with the alarm units located in the control room.

Test facilities are provided in the control room for injecting a test pressure to each transmitter.

7.2.2.3.4 Heat Transport Flow

Heat transport flow is measured in a total of twelve representative feeders, three in each direction through the reactor, in both coolant loops.

The orifice flow elements are installed in the horizontal run of the inlet feeder to the specified fuel channels. Flow transmitters are mounted on the three channelized instrument racks and the alarm units are located in the control room.

7.2.2.3.5 Logic Processing

Figure 7.2-2 shows a simplified block diagram of one channel of SDS 1.

There are three trip channels (D, E and F), having completely independent and physically separated power supplies, trip parameter sensors, instrumentation trip logic and annunciation. When any two of the three channels trip, the shutoff rods are dropped.

Because of the small number of trip parameters, general coincidence logic is used; that is, an entire channel trips when any parameter reaches its trip setting.

The trip system makes use of relay logic. Use of relays, in trip systems having few trip parameters leads to a very simple fail-safe design, capable of being completely tested during operation.

Use is being made of light-emitting diodes (LED's) in the shutoff rod trip network. No combination of open or short circuit diodes can disable a network from tripping safely. These diodes will be mounted on the SDS 1 panel and will indicate the correct operation of all contacts in the 2/3 networks when a particular trip parameter is tested, e.g. shutter test on the channel D ion chamber. The LED's will indicate correct operation of all contacts. In addition, on completion of a test, if a LED stays on, it indicates that a relay has not re-energized. This arrangement has been

selected to reduce the circuit testing time and to provide a positive indication of the state of the trip logic.

The shutoff rods are divided into two groups of fourteen. Each group is supplied with dual power supplies. Each clutch coil is held energized by the contact of a separate relay. The high volt-ampere rating of the clutch dictates this arrangement to ensure no relay contact overrating. Production reliability considerations (one failed relay results in only one rod dropping into the reactor) also dictate separate clutch relays.

7.2.2.3.6 Shutoff Rods

The mechanical construction and location of the 28 shutoff rods is discussed in Section 4.2.3. The withdrawal of shutoff rods is a control function and is described in the description of the reactor regulating system. The dropping of the shutoff rods by release of the direct current clutches is the function of SDS 1. The motor and drive circuits are completely independent of the clutch circuits, ensuring effective separation of the control and protective functions.

A partial drop test facility is provided in the clutch circuit to check the correct operation of each shutoff rod. There is a timer relay for each shutoff rod which will be set during commissioning to de-energize the clutch relay for sufficient time to allow the rod to drop a defined distance. The position of the rod after the test will be recorded to determine if there has been any deterioration in the clutch or rod performance.

7.2.2.3.7 Equipment Layout

The shutoff rod housings are located on the top deck plate where the clutches, motors, potentiometers, gearboxes and winches are accessible during shutdown.

There are separate cables and junction boxes for the clutch circuits and the motor drive circuits, to maintain separation between the regulating and protective channels.

The cubicles containing the group A and B shutoff rod drive logic, and the group A and B shutoff rod clutch trip logic and associated power supplies are located in the control equipment room behind the control room.

There is also a separate enclosed cubicle for each channel of SDS 1 located in the control equipment room. These channel trip cubicles contain the remaining logic.

In the main control room a separate main control panel bay is allocated solely to SDS 1. On it are mounted all the alarms, the trip and indicating meters, the test switches and buttons, the light-emitting diodes, the manual drive and test drop handswitches for the shutoff rods, and the manual trip button.

All of the reactor measuring devices on SDS 1 are field mounted in a manner which minimizes the possibility of common-mode failures with devices used for shutdown system two and for the regulating system. Their connecting cables are separately routed back to the three SDS 1 channel cubicles in the control equipment room.

7.2.2.3.8 Power Supplies

Three separately channeled Class II power supplies are provided to supply each channel of SDS 1. The logic is arranged so that any loss of power to a channel results in a channel trip.

The power supplies to the neutron instrumentation and the process transmitters are similarly arranged so that a channel will trip on loss of power due to a blown fuse or general loss of Class II power. Other types of internal instrument failures are less probable and are arranged to alarm in the control room.

7.2.2.4 Initiating Systems and Circuits for SDS 2

7.2.2.4.1 Horizontal Ion Chambers

Three uncompensated ion chambers, located in separate housings attached to the side of the calandria are provided for SDS 2. Each housing contains two separate ion chamber cavities and a test shutter. The second cavity will contain a spare ion chamber for SDS 2 since there is no access to the assemblies during operation. About five inches of lead shielding is provided between the inner end of the ion chambers and the outside of the calandria. The outer ends of the ion chamber tubes extend through the side wall of the shield tank where they are sealed by a bellows. Cooling is provided by the shield tank water.

The test shutter is the conventional piston-operated boral sleeve and will have the capability of increasing the flux at the SDS 2 ion chambers by up to 50%, and generating neutron rates up to 30% per second. The shutter test is initiated from the control room.

The output current from each ion chamber goes to an amplifier which will produce log neutron power, linear neutron power and rate log signals. Only the rate log output is utilized.

The design of the SDS 2 ion chambers and amplifiers will be different from that for SDS 1 and they will be provided by different suppliers to minimize the possibility of cross-linked faults.

7.2.2.4.2 Horizontal Flux Monitors

Three horizontal in-core flux monitoring assemblies, entering the side of the reactor, are provided for SDS 2. Each channel of SDS 2 is allocated a separate assembly, comprising three separate platinum detectors, of which two are spare. The assemblies are mounted in dry, sealed, guide tubes containing an inert gas.

The detectors are a double-ended design which permits the injection of a test current through the detector to check for continuity. They are linearly mounted so that the active length spans nearly ten feet in the reactor. This results in a signal which more truly reflects the average flux in the reactor.

The detectors are connected to linear amplifiers with a range of 5 to 150% of full power.

A test facility is provided to check each trip circuit from the control room by injection of a test current through the detector to the amplifier.

7.2.2.4.3 Logic Processing

There are three independent channels, G, H and J, from sensor location through the trip relays to the pair of quick-opening helium valves (see

Block Diagram Figure 7.2-3). General coincidence is used; that is either a high N or log N rate trip on any two of three channels will initiate poison injection. Fewer parameters, a 2/3 poison valve configuration, and low wattage requirements of the solenoid valves have permitted a very simple and reliable design with the parameter trip relays de-energizing the solenoids directly.

Generally, the circuit is very simple and considerably different in design from that of SDS 1. Both shutdown systems use fast-acting relays as logic devices although the relays will be provided by different manufacturers.

One manual trip button is provided for each channel in the control room. A local test trip button for each channel is mounted on each SDS 2 channel cubicle, which tests the trip seal-in relay. Interlocks are provided so that a channel may not be tested if any other channel is in the trip state. This helps to avoid spurious poison injection.

There are two poison injection valves and an interspace vent valve associated with each SDS 2 channel. Special test circuit logic blocks, for approximately 600 seconds, the testing of the next SDS 2 channel, so that the helium pressure which built up in the interspace from the previous test has bled away through the vent valve before the associated downstream valve is opened on test. Use is made in SDS 2 of electro-mechanical indicators for valve position and channel test state.

The mechanical components of the poison injection system are described in Section 4.6.

7.2.2.4.4 Equipment Layout

The helium storage tanks, quick-opening valves, and poison tanks are located outside the containment where they are accessible during reactor operation.

The ion chambers and in-core detectors for SDS 2 are horizontally mounted and their cables are routed in conduit through the side of the containment to ensure complete separation from SDS 1.

The cubicles containing the logic for each channel are located in three channelized instrument rooms in the Reactor Building. Since local coincidence is not used, there are no cross connections between each channel's trip logic from sensor to valves. The reactor trip comparators are mounted in separate instrument cubicles in these rooms so that the trip setpoints are not accessible to the operator in the control room. The design incorporates safeguards to ensure that access to these cubicles is available only under appropriate administrative control.

A separate panel in the main control room is allocated solely to SDS 2. On it are mounted all the alarms, the trip test and channel select switches, the electro-mechanical indicators, the manual trip button, and indicating meters. The latter are buffered from the trip instrumentation and provide an indication only of the neutron instrument signals.

7.2.2.4.5 Power Supplies

A separately channeled Class II power supply is connected to each of the SDS 2 channels. The logic and instrumentation have been designed so that a channel will trip on loss of power due to a loss of Class II supply or blown fuse.

7.2.3 Design Bases

The design and installation of SDS 1 and SDS 2 will be in General Agreement with the following specifications:

- IEEE 279 Nuclear Power Plant Protection Systems
- IEEE 308 Class IE Electrical Systems for Nuclear Power Generating Stations
- IEEE 323 General Guide for Qualifying Class I Electrical Equipment for Nuclear Power Generating Stations
- IEEE 338 Criteria for the Periodic Testing of Nuclear Power Generating Protective Systems
- IEEE 344 Trial Use Standard, Guide for Seismic Qualifications of Class I Electrical Equipment for Nuclear Power Generating Stations

SDS 1 and SDS 2 are independent in design and operation and are free from operational connection with any of the process systems, including the regulating system.

The provision of two reactor shutdown systems permits the assumption that at least one will operate following any single process failure for which both are designed to be effective.

The design philosophy being applied to the two shutdown systems is to keep them functionally and geometrically independent of each other and functionally independent of the station regulating systems.

The functional independence is basically achieved by the adoption of two different shutdown principles. The first shutdown system uses mechanical shutoff rods while the second shutdown system employs liquid poison injection into the moderator.

The geometric independence is achieved in several ways. The mechanical shutoff rods have a poised position just above the calandria and enter the core from the top. The poison injection nozzles enter the calandria from the side in a horizontal direction.

The process equipment associated with the liquid poison injection is to the side of the reactor, whereas the mechanisms for the mechanical shutoff rods are on the reactivity mechanisms deck above the reactor.

Neutron measurements for the first shutdown system are made from the top and one side of the reactor, while neutron measurements for the second shutdown system are made from the other side of the reactor.

7.2.4 Safety Evaluation

7.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Reactor Trip System:

See Section 7.2 of Appendix 7-A of this report for a listing of the applicable General Design Criteria.

7.2.4.2 Assessment

Information is presently incomplete for evaluating the design of the Reactor Trip Systems with regards to meeting the following Standards:

IEEE 317 Electric Penetration Assemblies

IEEE 379 Guide for Single Failure Criterion

IEEE 384 Criteria for Separation of Class IE Equipment and Circuits

There is insufficient information for evaluating conformance of the testing method for the Reactor Trip System to Regulatory Guide 22. In accordance with

the NRC's standard review procedure, periodic testing of actuation devices (e.g. solenoids, breakers) is required. These actuation devices of the Reactor Trip System must be periodically tested when the reactor is in operation.

The Reactor Trip System must, as the Acceptance Criteria for Controls (see Section 7.2, Appendix 7A) indicate, be designed in accordance with General Design Criterion 22. This criterion states that diversity shall be used to prevent the loss of protective function. Diversity can be achieved by using either equipment diversity or functional diversity. Equipment diversity is incorporated into a system by using different equipment to measure the same variable, make the same logic decision, or carry out the same protective action. Functional diversity can be achieved by using more than one variable to detect a particular accident condition.

The Reactor Trip System of a typical PWR (as licensed for U.S. operation) senses numerous system variables - to provide functional diversity. A comparison has been made of the trip parameters for the CANDU, as listed in Tables 7.2-1 and 7.2-2 with the Reactor Trips as used on PWR's. In order to achieve a functional diversity that is comparable to the one incorporated in PWR's, the trip parameters listed in Tables 7.2-1 and 7.2-2 would have to be supplemented with:

- Overtemperature ΔT
- Overpower ΔT
- Pressurizer Low Pressure
- Pressurizer High Pressure
- Pressurizer High Water Level
- Reactor Coolant Pump Bus Underpower

- Reactor Coolant Pump Bus Underfrequency
- Low Feedwater Flow

7.2.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

Provide functional diversity of the Reactor Trip System as discussed in Section 7.2.4 above.

TRIP PARAMETER	DETECTOR TYPE	SETPOINT (TENTATIVE)	INSTRUMENT* DELAY TIME (TENTATIVE) SECONDS	CONDITIONING PARAMETERS
High Neutron Power	Vertical In-core Detectors	115% Full Power	0.15	-
High H.T.** Pressure	Pressure Trans- mitters	10.58 MPa(g) (1549 psia)	0.21	
Low Gross Coolant Flow	ΔP Trans- mitters	(55% Nominal)	0.40	Log Power > 2% Full Power
High Rate Neutron Power	East Ion Chambers	10%/Sec	0.25	
Access Control Key Interlock	-	-	-	-
Manual	-	-	-	-
Start-up [†] Count Rate				

* Does not include time for parameter to reach trip value

** H.T. - Heat Transport System

† For initial start-up only

TABLE 7.2-1(a)
SHUTDOWN SYSTEM NO. 1 TRIP PARAMETERS

TABLE 7.2-1(b)

SDS1 TRIP PARAMETER TABLE

SINGLE PROCESS FAILURE	PRIMARY TRIP PARAMETER	BACKUP TRIP PARAMETER
Loss of Regulation From High Power:		
Fast	Hi Neutron Power	Hi H.T. Pressure
Intermediate	Hi Neutron Power	Hi H.T. Pressure
Slow	Hi Neutron Power	Hi H.T. Pressure/Manual
Loss of Coolant:		
Large	Hi Neutron Power	Low Gross Flow/ Hi N Log Rate
Intermediate	Hi Neutron Power	Low Gross Flow
Small	Low Gross Flow	Manual
Loss of Class IV	Hi H.T. Pressure	Low Gross Flow
Loss of Regulation From Low Power:		
H.T. System Pressurized	Hi N Log Rate	Hi H.T. Pressure
H. T. System Depressurized	Hi N Log Rate	Hi Neutron Power

H. T. - Heat transport system

TABLE 7.2-2(a)

SECOND SHUTDOWN SYSTEM

TRIP PARAMETER CHARACTERISTICS

TRIP PARAMETERS	DETECTOR TYPE	SETPOINT (TENTATIVE)	INSTR.* DELAY TIME (TENTATIVE) SECONDS	CONDITIONING PARAMETERS
Hi Neutron Power	Horizontal In-Core Detector	130%	0.10	-
Hi Log N Rate	Horizontal Ion Chambers	25%/Sec	0.20	-
Manual	-	-	-	-

* Does not include time for parameter to reach trip level

TABLE 7.2-2(b)

SECOND SHUTDOWN SYSTEM

TRIP PARAMETER TABLE

DUAL ACCIDENTS	PRIMARY TRIP PARAMETER	BACKUP TRIP PARAMETER
Loss of Regulation + SDS1 Failure	Hi Neutron Power	Hi Log N Rate
Large Loss of Coolant + SDS1 Failure	Hi Neutron Power	Hi Log N Rate

FIGURE 7.2-1 SHUTDOWN SYSTEM NO. 1 & NO. 2 TRIP PARAMETERS

FIGURE 7.2-2 SHUTDOWN SYSTEM NO. 1 - BLOCK DIAGRAM

FIGURE 7.2-3 SHUTDOWN SYSTEM NO. 2 - BLOCK DIAGRAM

7.3 Engineered Safety Feature Systems

7.3.1 Function

Adequate instrumentation and controls must be provided to sense accident situations and to initiate the operation of necessary Engineered Safety Features. The occurrence of a limiting fault, such as for example a steam-line break or a loss-of-coolant accident, requires a reactor trip and the actuation of one or more Engineered Safety Features (ESF) in order to prevent or mitigate damage to the core and Reactor Coolant System components and to insure containment integrity.

7.3.2 Description

The plant contains the following safety feature systems:

- Containment System
- Emergency Core Cooling System

These systems are independent in design and operation and free from operational connections with any of the process systems, including the regulating system.

7.3.2.1 Instrumentation and Logic

Each process and nuclear measurement loop essential for the operation of a safety system is triplicated such that a single loop component of power supply failure will not incapacitate or spuriously invoke the operation of the safety system. The design approach emphasizes isolation between loops of different channels and between the different safety systems. This is achieved by the use of unique transmitter mounting racks, electrical cabinets and power supplies for each channel.

The design aim is that instrumentation and logic be such that each trip parameter path is testable right from the primary transducer to the final release circuit. Two out of three release logic allows this practice.

7.3.2.2 Equipment Layout and Cable Routing

Emphasis has been placed on the independence between the two groups (comprising both safety and process systems), and retaining separation between channels of systems. However, the same three routes are used for all systems in a group, i.e., the coupled channel approach. Except for this common routing, all other aspects of independence between safety and process systems are followed.

Group one comprises approximately 95% of the total instrumentation and power requirements. Group two comprises the other 5 percent.

The group one control area is the main control room. Cables exit the reactor building on one side and run to the control equipment room.

The secondary control area, which houses the group two controls and instrumentation, is in the basement of the reactor auxiliary bay. Cables for this group leave the reactor building sub-grade on a different side than the group one cables. The group two controls and instrumentation are suitably buffered and retransmitted to the main control room so that centralized control can normally be exercised.

7.3.3 Design Bases

In order to protect the plant against common mode incidents such as fires, turbine missiles, and aircraft strikes that could affect many safety and safety related systems at the same time, all systems have been divided into two groups. These two groups are physically separated, so that no common mode incident occurring either inside or outside the reactor building could disable more than one of these groups.

The ultimate requirements are to avoid exceeding the siting release limits, but the actual design requirements for the two groups are to:

- (a) shutdown the reactor,
- (b) remove decay heat,
- (c) prevent any subsequent process failures,
- (d) supply necessary information to assess the state of the nuclear steam supply.

7.3.3.1 Group One Systems

Group one systems comprise the normal plant operating systems and the first line of defense safety systems. More specifically the systems are:

- (a) reactor regulating system,
- (b) shutdown system No. 1, (SDS 1)
- (c) emergency core cooling
- (d) all process systems except auxiliary moderator cooling.

7.3.3.2 Group Two Systems

Group two systems comprise systems that, acting together, will be capable of independent reactor shutdown, containment isolation, and decay heat removal from the ultimate heat sink. In particular, these systems are:

- (a) Shutdown System No. 2 (SDS 2)
- (b) containment (including dousing, isolation, etc.)
- (c) auxiliary moderator cooling
- (d) decay heat removal.

7.3.4 Safety Evaluation

7.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Engineered Safety Feature Systems:

See Section 7.3 of Appendix 7-A of this report for a listing of the applicable General Design Criteria.

7.3.4.2 Assessment

We do not have sufficient information for evaluating the conformance of CANDU's Engineered Safety Feature Actuation System (ESFAS) to all the IEEE Standards listed in The Appendix 7-A.

The information available for review indicates that an actuation system is provided which is separate from the Instrumentation and Control System used for normal operation, but it is unclear as to what signals are required to initiate specific safety features.

Before going into details, it should be stressed that the design philosophy for the ESFAS as used for the CANDU is quite dissimilar from the design philosophy as used for PWR's in the U.S.

The protection philosophy for the CANDU uses two groups of systems:

- Group 1 systems comprises systems in the first line of defense.
- Group 2 systems comprises systems in the second line of defense.

It should be mentioned that Group 1 systems of the CANDU comprise the normal plant operating systems. This philosophy of combining safety systems and control systems is not in agreement with General Design Criterion 24 which requires:

"The protection system shall be separated from control systems to the extent that failure of any single control system component or channel which is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy and independence requirements of the protection system."

Typical Engineered Safety Feature Systems that have not been incorporated into the CANDU's protection philosophy are:

- Containment and Reactor Vessel Isolation System
- Containment Heat Removal and Depressurization System
- Auxiliary Feedwater System
- Containment Air Purification and Cleanup System
- Containment Gas Control System

General Design Criterion 20 implies that automatic initiation is required for all above-mentioned protective functions.

To assure that the safety functions of the ESFAS can be performed when needed, it is, according to Design Criterion 20, necessary to incorporate in-service testability for all engineered safety features. In the case of some engineered safety feature systems, testing of certain protective functions may disrupt reactor operation. In such cases, acceptable methods for including the actuation devices in periodic tests of the protective system are:

- (a) Testing the actuation devices and the actuated equipment individually or in selected groups.
- (b) Preventing the operation of certain actuated equipment during a test of the actuation devices.
- (c) Designing the system such that operation of the actuated equipment requires the operation of more than one actuation device for actuation devices being individually tested.

In-service testability of the two groups of safety-related systems for the CANDU, according to presently available information, does not include features for testing actuation devices during normal plant operation.

7.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- (a) Additional plant parameters should be monitored which are indicative of whether or not predetermined safety limits are exceeded.
- (b) An Engineered Safety Feature Actuation System be added consisting of two discreet portions of circuitry:
 - 1. An analog portion consisting of three redundant channels per parameter or variable to monitor various plant parameters.
 - 2. A digital portion consisting of two redundant logic trains which receive inputs from the analog protection channels and perform the logic operations required to actuate the

Engineered Safety Features. The digital portion shall also incorporate the capability for sequencing the power demands on the Engineered Safety Feature busses (supplied by either preferred or standby power supply).

7.4 Systems Required For Safe Shutdown

7.4.1 Function

Parts of the nuclear steam supply system are required to achieve and maintain a safe shutdown condition of the plant. These systems are required to:

- Prevent the reactor from achieving criticality
- Provide an adequate heat sink such that design and safety limits are not exceeded.

Examples of systems required for achieving and maintaining a safe shutdown are the auxiliary feedwater system, and the residual heat removal system.

Systems required for safe shutdown must be equipped with adequate instrumentation as well as with the necessary automatic and manual control devices required to reach and maintain a hot shutdown with provisions to go subsequently to a cold shutdown.

7.4.2 Description

It is always necessary to shutdown the reactor, when initiating a cooldown procedure. Cooldown of the heat transport system is a function of the boiler pressure program, which ramps the boiler pressure setpoint down to give the constant rate of change requested by the operator.

A Reactor Trip is detected by the computer control programs and the following actions are initiated:

- drop mechanical control absorbers
- fill light water control absorbers
- change control mode to low-power, fixed reactor setpoint.

The generator remains connected to the grid system on a reactor trip. The governor valves are closed either by the computer or via suitable hardware logic.

In case the control room becomes uninhabitable, enough display and control instrumentation is provided at a location remote from the control room (the Secondary Control Area) to allow the plant to be shut down and maintained in a safe shutdown condition.

7.4.3 Design Bases

The Digital Computer System is the key control element for systems required for safe shutdown. The Design Bases for the Digital Computer System are summarized in Section 7.7.3.1.

Additional information, such as: Control-Hardware-Specifications, Process and Instrumentation Diagrams (P&ID's), Logic Diagrams, Layout Drawings would be required for a more detailed evaluation of the design bases used for systems that are required for safe shutdown.

7.4.4 Safety Evaluation

7.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Systems Required for Safe Shutdown:

See Section 7.4 of Appendix 7-A of this report for a listing of the applicable General Design Criteria.

7.4.4.2 Assessment

Systems required for safe shutdown must meet the requirements of General Design Criteria 2. The Digital Computer System, which is the key control element for these systems, must meet the requirements of this Criterion.

The limited information available indicates that the Digital Computer System is not designed for operation during and after certain design bases events such as earthquakes, accidents, etc.

A review of the descriptive information for the computer indicates (see Section 7.7.2.1) that the computer architecture does not satisfy the design criteria for physical separation.

7.4.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

Systems required for safe shutdown be controlled by hardware which is separate from the Digital Computer System described in Section 7.7. The proposed control architecture has the following advantages:

- Only the control hardware for systems that are required for safe shutdown must meet the requirements of General Design Criterion 2.
- Providing different Control Hardware for Systems required for safe shutdown and for Systems not required for safety reduces the complexity of the control structure.
- Reduced Complexity of the control hardware for systems required for safety leads to increased reliability.

7.5 Safety Related Display Instrumentation

7.5.1 Function

Safety-related display instrumentation comprises those indicators that provide the operator with vital information to aid in the safe shutdown of the plant, to maintain the plant in a safe hot shutdown condition and to remove decay heat after postulated incidents such as LOCA.

7.5.2 Description

The control room instrumentation is based on the philosophy of having sufficient information displayed to allow the unit to be controlled safely from the control room.

To achieve this goal, all indications and control essential for operation (start-up, shutdown and normal) are located on the control room panels. Also located there are controls for any system requiring attention within 15 minutes of an alarm occurrence. For systems not requiring attention within 15 minutes, local control is provided.

Most information is presented to the operator via the station computer system described in Section 7.7. However, sufficient conventional display, annunciation and recording of plant variables is included to allow the plant to be properly run in the shutdown condition with both computers out of service.

Special message windows are illuminated independently of the computers for all alarm conditions which cause reactor trips, power runbacks, turbine-generator trips, high voltage breaker trips, and some other important system alarms. These windows are located on the top section of the main control panels.

7.5.3 Design Bases

Alarm conditions which cause reactor trips and some other important system alarms are displayed via annunciators which are inserted in the top sections of the main control panels. The message windows of these annunciators are illuminated independently of the computers. Safety related information is presented to the operator by conventional display of plant variables. This conventional display of vital plant variables is supplemented by a computer based information system, the design bases for which are described in Section 7.7.

The flow and the temperature at the outlet of each Reactor Fuel Channel are measured. There is presently no information as to where and how these variables will be displayed.

7.5.4 Safety Evaluation

7.5.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Safety Related Display Instrumentation:

See Section 7.5 of Appendix 7-A of this report for a listing of the applicable General Design Criteria:

7.5.4.2 Assessment

General

Insufficient information was available to evaluate compliance of safety-related display instrumentation to all regulatory requirements. To perform a detailed evaluation, additional information is needed, such as: equipment specifications, lay-out diagrams, description of the control rod position

indicating system, listing of variables to be displayed via "conventional displays", listing of messages or conditions to be displayed via annunciators.

The conceptual design of the safety-related Display Instrumentation is not in accordance with the NRC's acceptance criteria because:

1. Redundancy has not been implemented.
2. Independence from offsite power is not provided.
3. Design includes no provisions for periodic testing to assure availability.

For these reasons, the Safety-related Display Instrumentation System is not licensable in the USA.

Reactor Fuel Channels

General Design Criteria 13 states "Instrumentation shall be provided to monitor . . . those variables that can affect . . . the reactor coolant pressure boundary . . . Appropriate controls shall be provided to maintain these variables and systems within prescribed ranges". A literal interpretation would indicate that relevant variables for each Reactor Fuel Channel must be monitored. Flow and Temperature at the outlet of each loop are two variables that would be indicative for potential problems. However, such an interpretation is not presently in effect.

7.5.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

To meet the NRC's criteria for Safety-Related Display Instrumentation, it will be necessary to change the conceptual design for the information

readouts and indications provided to the operator for monitoring conditions in the reactor, the reactor coolant system, the containment including all safety features. The recommended modifications are:

1. All monitoring channels shall be redundant, to assure that wrong indication due to device malfunction will not cause false action or inaction by the part of the operator.
2. Redundant channels of indicating instrumentation should be isolated physically and electrically to assure that a single failure will not result in complete loss of information about a monitored variable.
3. Capability should be provided for checking, with a high degree of confidence, the operational availability of each system input sensor during reactor operation.
4. An indication system should be provided covering bypassed or deliberately inoperable conditions of safety systems.
5. The seismic design of the Safety-Related Display Instrumentation System should be in accordance with the applicable design criteria.

7.6 All Other Instrumentation Systems Required for Safety

7.6.1 Function

In addition to the previously described instrumentation systems required for safety (Reactor Trip, Engineered Safety Features, Safe Shutdown, Safety Related Display Instrumentation) other Instrumentation systems are required.

These systems include:

- Refueling interlocks (required to prevent or mitigate the consequences of refueling accidents)
- Steamline break - Instrumentation & Control (to initiate isolation of steam generators)
- Containment atmospheric monitoring (to provide operator with information needed to control hydrogen and oxygen gases following postulated accidents)

7.6.2 Description

Only the refueling interlocks and related controls are described subsequently. Steamline break - Instrumentation and Containment atmospheric monitoring has not been incorporated into the existing design.

The control system for the fuel handling system consists of one of the station's computer-controllers (for details, see Sections 7.7.2.1 & 7.7.2.2) and, in addition, includes consoles and control panels specific to the fuel handling system.

The requirements of the fuel handling control system are:

- a. To control, remotely and reliably, all operations of the fueling machine and its associated systems and services, the new fuel loading system and the spent fuel transfer system.

- b. To exercise control through a pre-programmed computerized system and to permit manual intervention by the operator, if desired.
- c. To provide a printed record (log) of fuel handling operations, on demand
- d. To permit operation by personnel who, prior to their operational training, had no previous experience with computer-controlled machines.

Interconnections between the control system components and the fuel handling equipment are shown in the block diagram, Figure 7.6-1.

The operator has a choice of four modes of control: automatic sequence, automatic single-step, manual operation of individual outputs through the protective system, and direct manual operation of individual outputs.

The control console contains an automatic control panel, a cathode ray tube (CRT) display, a keyboard and a printer. Also provided are facilities for producing a printed log of operations. The operator performs all normal control functions from the automatic panel and the keyboard, and views the values of selected machine variables and the current log of operations on the CRT display. The printer provides a printed copy of the log, if requested. Through the keyboard the operator is able to introduce data into the system.

A simulator associated with the fuel handling controller is provided in order to check out control system operation without actually operating the mechanical equipment. Each output device or drive can be controlled manually. This facility is useful during commissioning and testing.

7.6.3 Design Bases

Refueling is controlled either automatically via a pre-programmed computerized system or by manual control. Command signals from the Computer Controller are (see Figure 7.6-1) interlocked via a protective System, but manual control may override the protective System. All information display is via the computer controlled CRT display.

Information presently available to us is very incomplete. To perform a more detailed analysis, additional information is required such as, for example:

- Protective System and the design bases for this system, including a description of interlocks.
- Type of measurements provided, if any, to assure that pressure tight seal is formed after the snout clamping mechanisms latch the fueling machine head on to an end fitting.
- Type of sensors provided, if any, for sensing the passage of fuel bundles and shield plugs into the machine.
- Type of position feedback obtained to assure that fueling machine head is in adequate alignment with an end fitting before the head clamps on it.
- Type of actions initiated if the Fueling Machine is jammed while operating in the automatic mode.
- Type of torque limiting, if any, when fueling machine removes the closure.
- Interlocks used, if any, to prevent damage to fuel channel due to too high torque applied or in case of mis-alignment.

7.6.4 Safety Evaluation

7.6.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of All Other Instrumentation Systems Required for Safety:

See Section 7.6 of Appendix 7-A of this report for a listing of the applicable General Design Criteria.

7.6.4.2 Assessment

Fuel Handling Control System

There is presently no nuclear power plant in commercial operation in the USA licensed to perform refueling operations while operating at power. Therefore, only inconclusive assumptions are used herein about licensing criteria that may be applicable to the control and instrumentation systems, as required for an on-line Refueling Machine. Existing licensing rules have, to a great extent, evolved out of the requirements for the present generation of PWRs and BWRs. Refueling interlocks are, according to the present Standard Review Plan, reviewed as part of the Systems Required for Safety. It is most probable that existing acceptance criteria for safety related systems will be invoked to the greatest extent possible.

This would mean the "Protective System" as shown in Figure 7.6-1 would have to satisfy the following requirements:

- Meet the single failure criterion
- Have the required redundancy
- Capable of functioning during and after certain design basis events

It should be noted that information required for safe operation of the refueling machine during normal and accident conditions would also have to meet the above listed requirements. Indication systems would have to be incorporated to indicate bypassed or deliberately inoperable conditions of safety systems (Example: In case of manual intervention by the operator). We believe that interlocks will also be required to guard against operator error as well as a malfunctioning computer controller.

7.6.4.2.1 Steamline Break - Instrumentation & Control

In the event of a steamline break, it is necessary that a steam generator repressurize in order to shut down the plant in an orderly and controlled fashion. To meet the single failure criterion of 10 CFR 50 Appendix A, it is necessary to provide steam line break instrumentation and controls consisting of redundant, seismically qualified channels. Each channel must, independently, be capable of initiating the required protective actions. Steam pressure must be monitored for each channel by pressure transmitters located in the outlet line of each steam generator. When a pre-determined low pressure setpoint is reached, closure of the steam and feedwater isolation valve shall be automatically initiated. The Steamline Break interlock circuit must include provisions for manually bypassing steam generator isolation during the startup or shutdown mode when steam generator outlet pressure is low. Such a bypassed condition must be alarmed in the control room.

7.6.4.2.2 Containment Atmospheric Monitoring

The containment atmospheric monitoring system is needed in support of the systems required by General Design Criterion 41. The containment atmospheric

monitoring system must be designed in accordance with all applicable design criteria listed in Appendix 7-A of this report.

7.6.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

1. Control and Instrumentation requirements for the Fuel Handling Control System must be reviewed with regard to meeting: The single failure criterion, the redundancy requirements, the requirement for functioning during and after design basis event. The entire system shall be described in detail with special emphasis given to component malfunctions or operational errors during refueling. The requirements for adjustable overload and lowload interlocks shall be discussed. Features shall be provided for preventing fuel assembly damage and potential release of fission products due to operator error or electric or hydraulic failure. The "Protective System" (see Figure 7.6-1) should be designed in accordance with the requirements outlined in Section 7.6.4.1.
2. Steamline break instrumentation must be added to satisfy the safety requirements as outlined in Section 7.6.4.2.
3. Containment Atmospheric Monitoring System must be added to satisfy General Design Criterion 41.

FIGURE 7.6-1 FUEL HANDLING CONTROL BLOCK DIAGRAM

7.7 Control Systems Not Required For Safety

7.7.1 Function

Systems described in this Section perform the following functions:

- Establish and maintain power equilibrium between the primary and secondary system during steady state operating conditions.
- Constrain operational transients in case of rapid load changes or system upsets.
- Provide the Power Plant operator with information indicating the range of required input and output control parameters.
- Provide the Power Plant operator with the capability for assuming manual control of certain Systems.

7.7.2 Description

7.7.2.1 Digital Computer System-Hardware

Digital computers are employed for station control, alarm annunciation and data display. Direct digital control is used for such functions as regulating reactor power, primary heat transport pressure, steam pressure and boiler level. The system consists of two independent digital computers with each computer being capable of controlling the unit.

Each computer is a modern high speed machine with its own private core memory, very reliable disc memory, unit input-output interface and peripheral equipment. The two computers communicate via a computer-to-computer data link. However, unit control by either computer is not dependent upon this data link. The plant input-output system is designed to be immune to most normal types of plant faults. For example, overvoltage (either AC or DC) on any input-output channel does not affect the operation of other input-output channels. The system is carefully organized so that maintenance on one computer can take place while the unit is being controlled by the other computer. A fault in any essential part of one computer results in automatic transfer of control to the other computer. The unit is automatically shutdown if both computers fail.

The high reliability of this dual computer control system results from combining highly reliable solid state hardware with a self-checking system. Faults, either software or hardware, are detected by a combination of internal hardware and software self-checking plus an external "watch-dog timer". Detection of a fault results in control being transferred to the other computer. A restart system, which automatically reloads the core

memory from the disc memory and restarts the computer, is combined with the fault detection to provide a system practically immune to transient faults.

The computer controllers are built on a modular basis with large sections being replaceable. This simplifies fault diagnosis and reduces computer downtime.

All functions essential to the operation of the plant are incorporated in both computers. Typical duplicated functions are:

- a. Reactor Power Control
- b. Plant Load Control
- c. Steam Pressure Control
- d. Alarm Annunciation
- e. CRT Data Display

Other functions, not essential to unit operation, may be resident in one computer only. Two such functions are:

- a. Fueling Machine Control Programs
- b. Turbine Run-Up Program

Keyboards mounted in the control room panels and on the operator's desk provide the means for operator communication with the computers.

A block diagram of the Digital Computer-Architecture is shown in Figure 7.7-1.

Typical control transfer interlock systems for duplicated analog and digital output circuits are shown in Figure 7.7-2. The control transfer system is such that if the master computer controller, DCCX, is operational it has control of the field devices. Referring to Figure 7.2-2, for the analog

circuit, DCCX actuates the A contacts and DCCY the B contact. With both computers operational, A1 and B are closed while A2 is open. Thus, the signal from DCCX is connected to the field device while that from DCCY is blocked by contact A2. If DCCX fails, e.g. the operation monitor times out A1 opens and A2 closes so isolating DCCX and allowing DCCY to take over control. If DCCY should also fail, all control is lost, contact B opens and the field device assumes a fail safe state. As soon as DCCX is re-started it automatically takes over control from DCCY.

For digital outputs, two control transfer schemes are used. Where a small number of contacts are actuated by a control program a form C contact is used in each output circuit to transfer control. These contacts are actuated by DCCX and when de-energized transfer control to DCCY. When a large number of contact outputs are used for a particular control function, a different approach is adopted to improve reliability. Three additional contacts in DCCX are wired as digital inputs into DCCY. DCCX closes these contacts when it is in control. The duplicated control program in DCCY performs a 2 out of 3 check on these inputs and maintains its own contact outputs in an open state as long as DCCX has control.

7.7.2.2 Digital Computer System - Software

All computer programs are written in Assembler-language for efficient utilization of computer time and memory. The programs can be broadly classified in the following two groups:

- Executive Program
- Functional Programs

The executive program resides in a protected part of core memory and forms the heart of the software system. The great majority of programs in the

second group reside on the disc memory and are transferred into a common area of core, the overlay area, for execution. However, functional programs which must be executed frequently, once a second or less, reside in core to avoid the time overhead involved in disc transfers.

The executive program is responsible for the time scheduling of all functional programs and also for checking their correct operation. A detected fault results in a program retry and if the retry is unsuccessful the failed program is automatically turned off to avoid jeopardizing the overall system operation. The function of a failed duplicated program is automatically taken over by the other computer. Each computer program is designed to restart with minimum disturbance to the plant. In the case of the master computer, DCCX, if two restarts occur within a short period of time the second restart is inhibited and control is transferred to the stand-by computer, DCCY. DCCX must be restarted by the operator. This approach is adopted to ensure that a hard fault in DCCX will not result in a continuous restart cycle and failure to transfer control.

The Functional Programs can be broadly classified into the following groups:

- Alarm, Data Logging and Display Programs (these programs generate the Alarm messages for the CRT's, Line Printers, General Purpose Displays, Data Logging).
- Control Programs (these are subdivided into "Duplicated" and "Non-Duplicated" Control Programs).

Most of the plant annunciation is carried out through the computers. The computers handle the contact alarm scanner data and monitor selected analog inputs for the purpose of alarm annunciation. Either computer may

deal with all of the alarms but only one is selected at any time to produce the alarm messages. Transfer to the other computer is automatic in the event of a failure.

The control programs are designed to be independent of each other, to be immune to single faulty inputs and to carry out checks on their outputs. These requirements are met by ensuring that each program reads in all the inputs it requires then rejects any which are outside limits or which fail predetermined checks. All essential inputs are duplicated or triplicated to provide redundancy with median values or averages normally being used for control. Each program can alarm on any abnormal condition and, if necessary, turn itself off leaving its outputs in a safe state. If the program is one of the duplicated functions, then control will be maintained by the standby computer.

For major control loops duplicated control programs are used. These programs are described in Section 7.7.2.3.

Some control functions require a large amount of computer memory but are not essential to the operation of the unit for a period of several days. For these functions, Non-Duplicated Control Programs are used. Past experience has shown that most computer faults are cleared up within minutes so these functions may be performed in only one of the control computers without any significant loss of availability.

Non-Duplicated Control Programs are:

- Turbine Run-Up Control Program
- Fueling Machine Control Program

The turbine run-up program performs a series of pre-start checks to see that the set is ready for run-up. The program then runs the turbine up to speed in a manner designed to achieve the shortest run-up time consistent with the safety of the turbine. The operator will be able to monitor the progress of the turbine run-up and intervene if necessary. If any of the checks within the program are failed the operator can override them via one of the keyboards and allow the program to proceed with the run-up.

Two fueling machines are used for changing reactor fuel. The Fueling Machine Control Program establishes the sequential operation of a large number of steps with each containing a small number of permissives and control actions. In addition to the sequencing control, direct digital control of various loops is provided such as the automatic positioning of all drives. Both fueling machines are controlled from DCCY only since the fueling operations can be delayed, or completed manually, if the computer is not available. Two CRT's and a printer provide the operators with the information as to the step number and actions being taken by the computer.

7.7.2.3 Digital Computer System - Regulating Programs

Figure 7.7-3 shows the Block Diagram for the overall plant control system. The major control loops in the plant and their basic functions are as follows:

- a. Overall Plant Control - coordinates reactor power level with the turbine generator load demands. The control is of the reactor-following-boiler type. That is, turbine load is set to the desired value and the reactor supplies this load by simply attempting to maintain boiler pressure constant.

b. Reactor Control - manipulates the reactivity control devices so as to maintain reactor flux power at a setpoint consistent with turbine load demands, and to maintain a desired flux shape. The Block Diagram of the Reactor Regulating System is shown in Figure 7.7-4. Reactor control also embodies slow and fast power cutback facilities (reactor power setback and stepback routines) to keep the plant parameters within safe limits during upset conditions. Figure 7.7-5 illustrates the setback and stepback parameters.

A stepback is similar to a reactor trip, with the exception that it is a gentler power reduction and may terminate at any intermediate power level. Stepback signals are tabulated in Table 7.7-2. On a stepback the control mode is also changed to the low-power fixed reactor setpoint mode. The turbine is unloaded as necessary by the boiler pressure control algorithm which is used whenever boiler pressure control is via the plant loads.

Reactor power setbacks are applied directly to the neutron power setpoint and therefore override other demands for reactor power changes. Specifically, the normal mode of boiler pressure control is interrupted during a setback; Boiler Pressure Control is switched to its alternate mode of constant reactor power setpoint mode. When the condition clears, the setpoint is returned to boiler pressure control, unless the setback has proceeded to its endpoint in which case reactor power is frozen at its reduced level by placing the reactor power setpoint under manual control. Setback signals are tabulated in Table 7.1-1.

- c. Boiler Pressure Control - manipulates the reactor flux power setpoint so as to keep boiler pressure constant. During turbine load rejections, this controller opens the bypass valves to the main condenser.
- d. Boiler Level Control - manipulates the feedwater valves in order to maintain the water level in the boiler at a setpoint dependent upon reactor power.
- e. Heat Transport Pressure Control - controls the surge tank steam bleed valves and heaters to maintain heat transport pressure constant.
- f. Surge Tank Level Control - controls D₂O feed and bleed valves to maintain the surge tank level at a power-dependent setpoint. The heat transport system is essentially constant inventory in that the surge tank level is programmed to accept the swell from zero power hot to any given power level.

All of the above major controls are implemented using direct digital control from the two control computers. Digital control was chosen for these functions because the systems are complex and nonlinear. The execution of these duplicated control programs is scheduled by the Executive program. For details see Section 7.7.2.2.

In addition to the above major control loops, there are many smaller loops implemented using standard analog control hardware.

7.7.2.4 Process System Instrumentation

The process instrumentation for the nuclear steam supply system is designed to provide reliable control and monitoring of the process plant. Sufficient

automation is provided to ensure safe, reliable and economical operation. Control stations and visual indicators associated with safe and reliable operation of the plant are located in the central control room. Certain other controls and indicators are located elsewhere in the plant.

Instrumentation identification by measurement and/or activating variable, and by function is provided and is based on ISA S5.1.

All instrumentation lines comply with the appropriate parts of Section III of the ASME Boiler and Pressure Vessel Code.

Provision is made for the operation of instrumentation and control circuitry independent of loss of plant normal power. Normal power is backed-up by standby AC power for all instrumentation. Where feasible, elements and circuits are designed to fail safe. For a final control element, fail safe is the positioning which provides the safest process condition.

Wiring and cabling practice segregates conductors carrying appreciable AC or DC power, or driving unsuppressed inductive switching loads from signal cables. Signals are amplified at or near their source whenever possible. Ground loops are avoided by ensuring that only one ground is allowed on a signal circuit.

Standard commercial instruments and materials of suitable types are used in general. Special materials, treatments, finishes and dust-tight and air-conditioned enclosures are employed where necessary. The minimum number of different types of instruments are used in order to achieve economy in maintenance.

Temperature sensors are generally of the platinum resistance type and where practicable are used with protecting wells. Control valves generally use pneumatic motors of the diaphragm type.

7.7.2.5 Control Room

The control room will contain a unit control panel, switchyard and electrical services panel, an operator's desk with two high speed line printers, and a fuel handling and fueling machine control console.

The control panels are arranged so as to provide part of the boundary walls of the control room (see Figure 7.7-6). With the degree of automation provided, the need for an operator to perform any function at the control panels continuously or even frequently is extremely rare, therefore, the main control panels are designed as stand-up panels with no console. An exception to this rule is the fueling machine console where despite a high degree of automation, manual intervention will sometimes be required. To reduce interference with the rest of the control room, this console will be located to one side of the room out of the main traffic flow.

The panels are laid out on a system basis with the controls for a specific system being located in one panel bay. Spacing between instruments is kept to an acceptable minimum in an attempt to achieve a compact display of information.

In laying out each system, consideration is given to the relative location of the controls based on process function and/or plant location. The more complex process systems are displayed using colored graphic lines to represent the flow paths. The electrical systems are displayed on mimic diagrams. There are seven CRT's with their associated keyboards located on various

process panels for system parameter and graphic displays as shown in Figure 7.7-7. Two additional CRT's are located on the fueling control console to display control operations.

Computer driven graphic CRT displays replace many of the meters and recorders normally found on conventional panels. Sufficient redundancy is built into the display system to ensure a very high availability comparable to the dual computer controller system itself. The use of computer driven displays results in less congested panels and allows easier correlation of information. For the convenience of the control room operators, a CRT is located in their desk which allows them to view computer driven graphics or alphanumeric displays of any important plant parameters. The operators are able to obtain copies of any CRT displays by pressing a pushbutton on the operator's desk.

The annunciation system consists of a main control room annunciator and a field panel annunciator. The main control room annunciation system consists of two computer driven CRT's for alarm message presentation, a facility designed to provide a printed record of all alarm conditions with enough information to enable them to be arranged in chronological order of their occurrence. In addition, there are annunciators located in the top section of the main control panels. These annunciator windows are illuminated independently of the computer and alarm conditions which cause a reactor trip, turbine trip and other important system alarms.

A total of twelve computer driven T.V. monitor-type CRT displays are mounted in the control room to present unit operating information to the operator. Eight 17 inch monitors have full alphanumeric and graphic capabilities and are used to display process trend information, special graphic information,

measurement groupings and to provide operator to computer interfacing displays. Two 23 inch monitors are used to display alarm messages for remote viewing. These ten T.V. monitors may be switched to either computer controller for updating. Two other 17 inch T.V. monitors are provided for the display of fueling machine information. These displays are updated from DCCY only since this computer is used for fuel handling control.

The monitors are driven from controllers which refresh the displays 30 times/second. The computers transmit data to the controllers only when a display is to be changed or updated.

7.7.2.6 Control Equipment and Plant Computer Rooms

The control equipment room contains the control equipment required for those systems being controlled from the main control room.

A separate room, which will house the station and fueling machine computers, will be located adjacent to the control equipment room. This room will be completely enclosed and will have its own heating and air conditioning system to ensure a relatively dust and dirt free environment for the computers.

The equipment is generally contained in enclosed cubicles. Access for cables is provided by a network of cable pans which is located directly below the control center. As a general rule, all cabling is brought through openings in the floor from the cable pan area below.

The equipment is laid out with respect to function and a high degree of separation will be maintained. This separation principle applies to both equipment placement and cable routes, and attempts to minimize the possibility of unsafe conditions arising out of fire or physical damage to equipment.

The equipment in the control equipment rooms with the exception of the safety systems and some other direct wired cubicles will be connected to the field equipment via the control distribution frame (CDF). The CDF will allow cross-connections to be made between cables from control centre equipment and field mounted equipment. The safety system cabling will incorporate separate termination cubicles and/or direct point to point cabling.

The control equipment rooms will be fully air-conditioned to provide a suitable environment for the installed equipment.

7.7.3 Design Bases

7.7.3.1 Digital Computer System

Two independent digital computers, with each computer being capable of controlling the unit, are used for:

- Regulating reactor power, primary heat transport pressure, steam pressure and boiler level
- Fueling Machine Control
- Controlling turbine run-up (to achieve shortest run-up time consistent with the safety of the turbine)
- Alarm annunciation and data display

A fault in any essential part of one computer results in an automatic transfer of control to the other computer. The unit is shut down, if both computers fail.

Control Functions which, for a period of several days, are not essential to the operation of the unit are performed by only one computer.

7.7.3.2 Control Room

The control room panels are arranged so as to provide part of the boundary walls of the control room. Panels are laid out on a system basis with the controls for a specific system being located in one panel bay. Spacing between instruments is kept at a minimum in an attempt to achieve a compact display of information. Computer driven CRT displays are used extensively. These displays are mounted on the panels. In addition, there is a CRT on the operator's desk. This display allows the operator to review any important plant parameter. CRT displays are used for both safety related and non-safety related information display. In addition to this computer-based information display system, provisions have been made for some conventional display instrumentation. An annunciator system is provided which is independent from the computer. It is used for important system alarms.

7.7.3.3 Process System Instrumentation

The process instrumentation for the nuclear steam supply system is designed to provide reliable control and monitoring of the process plant. All instrumentation lines comply with the appropriate parts of Section III of the ASME Boiler and Pressure Vessel Code. Where feasible, elements and circuits are designed for "fail safe" operation.

7.7.3.4 Control Equipment & Plant Computer Rooms

The equipment is generally contained in enclosed cubicles. A high degree of separation is maintained. Part of the boundary wall between the Control equipment and the Control Room is formed by the Control Panel. The Plant Computer Room is adjacent to the Control Equipment Room.

7.7.4 Safety Evaluation

7.7.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Control Systems not Required for Safety:

See Section 7.7 of Appendix 7-A of this report for a listing of the applicable General Design Criteria.

7.7.4.2 Assessment

General

The NRC's acceptance criteria for systems not required for safety are summarized in Appendix 7-A. Additional requirements are: Section 9.5.1 of the NRC's Standard Review Plan, applicable "National Fire Codes" as issued by the National Fire Protection Association (NFPA) to establish guidelines for specific plant control areas.

Additional information would be required to evaluate conformance of the control equipment to the applicable standards listed in Appendix 7-A.

Lay-out drawings of panels (especially for safety related Systems) would be required to analyze the arrangement, redundancy of components, Testing- and Bypass-Features.

Digital Computer System

Presently, there are no NRC Regulatory Guides relating to the application of digital computers. The use of digital computers is acceptable to the NRC. As an example, four digital minicomputers may be used in modern U.S. nuclear power plants to process the parameters needed for calculating DNBR and linear heat rate (kW/ft).

The NRC's standard review plan for Section 7.7 indicates that special emphasis will be placed when new design features are introduced. The centralized dual digital computer system as used for the CANDU constitutes such a new feature.

The computer system performs the following basic functions:

- Plant control
- Plant monitoring
- Alarm display

A plant shutdown is required if both computers fail. In such a case, the operator loses simultaneously:

- Control over the unit
- Important information about the unit's performance at a time when this information is urgently needed for assessing the status of the plant

The NRC's Standard Review Plan for Section 7.7 indicates that following aspects of non-safety related Systems will be reviewed: "The circuit to circuit failure modes of a single non-safety control system and their effect on the protection system and gross failure models of non-safety control systems and their functional effect on the Protection System". The NRC's Standard Review Plan also indicates that the non-safety related instrumentation should:

"... minimize challenges to the safety system".

CANDU's digital computer system uses fewer components for performing the plant control functions than conventional analog control systems. It could therefore be argued that the probability of control hardware related forced plant shutdowns is decreased. But, as the Block Diagram for the Digital Computer System (see Figure 7.7-1) indicates, the computer is not performing

only the plant control related operations. Additional tasks performed by the computer are: Line Printer Control, Display Control, Fueling Machine Control, etc. This means that many additional digital and analog signals are fed into the computer for other than plant control purposes. This increases the probability that wiring related problems may infiltrate into adjacent control systems. It is therefore difficult to argue that the digital computer control system, as used for the CANDU, would cause fewer forced plant shutdowns than a comparable conventional analog control system. When the dual computing structure of the CANDU fails, very severe corrective action may be required, because control over the unit as well as important information are lost simultaneously. The NRC may, therefore, conclude that the Computer Architecture as provided for the CANDU, does not "minimize challenges to the safety system" to the same extent as those systems which are used in the latest generation of U.S. nuclear power plants.

It should also be mentioned that certain safety related control and display functions are performed by the digital computer system (for details, refer to Section 7.5.5). Within the computer architecture, as used for the CANDU, it is not ready to separate systems which are required for safety, from those systems which are not required for safety.

It is mentioned (Section 7.7.2.2) that "each computer program is designed to restart with minimum disturbance to the plant". We think that the NRC will require a more precise definition of "minimum disturbance". We also believe that by using a distributed control system this "minimum disturbance" can be reduced, because in case of a failure within a part of the overall system, other hardware components which remain operational, will attempt to initiate

the necessary corrective actions. Additional advantages of such a distributed control system are:

- Plant control functions can be isolated from the monitoring and data display system (a computer malfunction will not lead to loss of control over the unit and to simultaneous loss of vital information).
- Safety related systems can be isolated from systems not required for safety.

In Section 7.7.2.3 it is mentioned that "... many smaller loops are implemented using standard analog control hardware". We do not know what functions are performed by these loops and what type of back-up is provided by these loops (if any) in case of a computer malfunction.

In Section 7.7.2.2, reference is made to a "Turbine Run Up" Control Program. We would like to mention that in modern U.S. power plants a separate computer is used for this purpose. Past experience has shown that this approach leads to a clearer definition of responsibilities. It is known that many turbine related problems are due to improper start-up or shutdown procedures. If the "Turbine Run Up" computer is not supplied by the Turbine Manufacturer, then an attempt may be made to "prove" that turbine related problems are caused by the computer which was supplied by others.

Control Room

To satisfy the requirements of APCSB 9.5-1, the following must be considered:

- The control room is essential to safe reactor operation. It must be protected against disabling fire damage and should be separated from other areas of the plant by floors, walls and roof having minimum

fire resistance ratings of three hours (it is not acceptable to use the control panel as a boundary wall as shown in Figure 7.7-6).

- Fire detection in the control room, cabinet and consoles should be provided by smoke and heat detectors in each area. Alarm and annunciation should be added in the control room for alarming fires in any plant area.
- Control room floors, floor ceiling structures and walls should be designed to a minimum three-hour fire rating. All penetrations should be air tight.
- The control room ventilation intake should be provided with smoke detection capability to automatically alarm locally and isolate the control room ventilation system to protect operators by preventing smoke from entering the control room. Manually operated venting of the control room shall be available.

Safety related displays shall be redundant and shall be isolated from non-safety related displays (see Section 7.5 for more detailed requirements).

Control Equipment & Plant Computer Rooms

To satisfy the requirements of APCSB 9.5-1, the following must be considered:

- Fire detection in the cabinets should be provided by smoke and heat detectors. Alarm and annunciation should be provided in the control room.
- Primary fire suppression in the cable spreading area should be an automatic water system.

- Redundant safety related cable divisions should be separated by three-hour fire rated walls.
- The plant computer room shall be separated from other areas of the plant by barriers having minimum three-hour fire resistance rating. Automatic fire detection should be provided to alarm and annunciate in the control room and locally. Manual hose stations and portable water and halon fire extinguishers should be provided.

The Plant Computer Room shall also be designed in accordance with the requirements of National Fire Code 75. (Separate storage room for records should be provided).

7.7.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

1. Separate hardware should be provided for performing plant control, Plant Monitoring and Alarm display functions.
2. Safety related control and display functions should be performed by separate components.
3. "Turbine Run Up" Control feature should be performed by separate hardware.
4. The Control Room, Control Equipment Room and the Plant Computer Room should be redesigned and additional features should be incorporated as outlined in Section 7.7.4.2.

5. Safety related displays in the control room shall be redundant and shall be isolated from non safety related displays.
6. Features should be added in the Control Equipment Room, as outlined in Section 7.7.4.2.
7. Features shall be added in the Control Room for actuating, monitoring and testing the two independent ESF systems.
8. Additional features outlined in Section 7.5.4.3 should be incorporated into the panel design.

TABLE 7.7-1

SETBACK SYSTEM SIGNALS

Condition	Setback Rate %/Second	End Point % F.P.
Zone Control System Failure (low H ₂ O Pressure)	0.25	50
Spatial Control Off Normal	0.05	25
Fuel Overrating (Flux Mapping)	0.1	50
High Boiler Pressure	0.5	10
Surge Tank High Level	0.1	2
Boiler Low Level	0.5	2
Manual	0.5	2

TABLE 7.7-2

STEPBACK SYSTEM SIGNALS

Condition	Rod Drop Type	New Demand Power, % F.P.
Reactor Trip	Full Drop	5
Very Low Surge Tank Level	Full Drop	5
Turbine Trip (conditioned By Reactor Power 70% F.P.)	Partial Drop	60
Main Coolant Pump Trip	Partial Drop	0.7 x Previous Demand Power
High Heat Transport Pressure	Partial Drop	0.7 x Previous Demand Power

FIGURE 7.7-1 GENERATING STATION DIGITAL COMPUTER CONTROLLER - BLOCK DIAGRAM

FIGURE 7.7-2 SIMPLIFIED SCHEMATIC OF COMPUTER TRANSFER SWITCHING

FIGURE 7.7-3 OVERALL PLANT CONTROL - BLOCK DIAGRAM

FIGURE 7.7-4 REACTOR REGULATING SYSTEM - BLOCK DIAGRAM

FIGURE 7.7-5 UNIT REGULATING SYSTEM SETBACK & STEPBACK PARAMETERS

FIGURE 7.7-6 CONTROL CENTRE LAYOUT

FIGURE 7.7-7 TYPICAL CONTROL PANEL LAYOUT

APPENDIX 7-A

STANDARD REVIEW PLAN

ACCEPTANCE CRITERIA FOR CONTROLS

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

TABLE 7-1
ACCEPTANCE CRITERIA FOR CONTROLS

Table 7-1 contains the acceptance criteria for the review plans of Chapter 7. These acceptance criteria include the applicable general design criteria, IEEE standards, regulatory guides, and branch technical positions (BTP) of the Electrical, Instrumentation and Control Systems Branch (EICSB). The table was prepared by EICSB for use by its members in reviewing Chapter 7 and for use by secondary review branch reviewers.

The applicability of these criteria to specific sections of Chapter 7 is indicated by an X in the matrix listing of criteria and SAR sections. There is a corresponding table (8-1) at the end of Chapter 8 covering the acceptance criteria of safety-related power supplies. The BTP listed in Tables 7-1 and 8-1 are contained in Appendix 7-A to the Chapter 7 review plans.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20558.

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ACCEPTANCE CRITERIA FOR CONTROLS - TABLE 7-1

CRITERIA	TITLE	APPLICABILITY (SAR Section)							REMARKS
		7.1	7.2	7.3	7.4	7.5	7.6	7.7	
1. 10 CFR Part 50									
a. 10 CFR §50.34	Contents of Application: Technical Information	X	X	X	X	X	X	X	
b. 10 CFR §50.36	Technical Specifications	X	X	X	X	X	X		
c. 10 CFR §50.55a	Codes and Standards	X	X	X	X	X	X	X	
2. General Design Criteria (GDC), Appendix A to 10 CFR Part 50									
a. GDC 1	Quality Standards and Records	X	X	X	X	X	X		
b. GDC 2	Design Bases for Protection Against Natural Phenomena	X	X	X	X	X	X		
c. GDC 3	Fire Protection	X	X	X	X	X	X		
d. GDC 4	Environmental and Missile Design Bases	X	X	X	X	X	X		
e. GDC 5	Sharing of Structures, Systems, and Components	X	X	X	X	X	X		
f. GDC 10	Reactor Design	X	X	X	X	X	X		
g. GDC 12	Suppression of Reactor Power Oscillations	X	X			X		X	
h. GDC 13	Instrumentation and Control	X	X	X	X	X	X	X	
i. GDC 15	Reactor Coolant System Design	X	X			X		X	
j. GDC 19	Control Room	X	X	X	X	X	X	X	
k. GDC 20	Protection System Functions	X	X	X	X	X	X		
l. GDC 21	Protection System Reliability and Testability	X	X	X	X	X	X		
m. GDC 22	Protection System Independence	X	X	X	X	X	X		

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section)							REMARK
		7.1	7.2	7.3	7.4	7.5	7.6	7.7	
n. GDC 23	Protection System Failure Modes	X	X	X	X	X	X		
o. GDC 24	Separation of Protection and Control Systems	X	X	X	X	X	X	X	
p. GDC 25	Protection System Requirements for Reactivity Control Malfunctions	X	X			X			
q. GDC 26	Reactivity Control System Redundancy and Capability	X	X		X	X		X	
r. GDC 27	Combined Reactivity Control Systems Capability	X	X		X	X		X	
s. GDC 28	Reactivity Limits	X	X			X	X	X	
t. GDC 29	Protection Against Anticipated Operational Occurrences	X	X	X	X	X	X	X	
u. GDC 33	Reactor Coolant Makeup	X			X	X	X		
v. GDC 34	Residual Heat Removal	X		X	X	X	X		
w. GDC 35	Emergency Core Cooling	X	X	X		X	X		
x. GDC 37	Testing of Emergency Core Cooling System	X	X	X		X	X		
y. GDC 38	Containment Heat Removal	X		X		X	X		
z. GDC 40	Testing of Containment Heat Removal System	X		X		X	X		
aa. GDC 41	Containment Atmosphere Cleanup	X		X		X	X		
bb. GDC 43	Testing of Containment Atmosphere Cleanup Systems	X		X		X	X		
cc. GDC 44	Cooling Water	X		X		X	X		
dd. GDC 46	Testing of Cooling Water System	X		X		X	X		
ee. GDC 50	Containment Design Basis	X				X	X		

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TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section)							REMARKS
		7.1	7.2	7.3	7.4	7.5	7.6	7.7	
ff. GDC 54	Piping Systems Penetrating Containment	X		X		X	X		
gg. GDC 55	Reactor Coolant Pressure Boundary Penetrating Containment	X		X		X	X		
hh. GDC 56	Primary Containment Isolation	X		X		X	X		
ii. GDC 57	Closed Systems Isolation Valves	X		X		X	X		
3. Institute of Electrical and Electronics Engineers (IEEE) Standards:									
a. IEEE Std 279-1971 (ANSI N42.7-1972)	Criteria for Protection Systems for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See 10 CFR §50.55a(h) and Reg. Guide 1.62
b. IEEE Std 308-1971	Criteria for Class IE Electric Systems for Nuclear Power Generating Stations	X			X	X	X		See Reg. Guide 1.32.
c. IEEE Std 317-1972	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.63.
d. IEEE Std 336-1971 (ANSI N45.2.4-1972)	Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.30.
e. IEEE Std 338-1971	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems	X	X	X	X	X	X		
f. IEEE Std 344-1971 (ANSI N41.7)	Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations	X	X	X	X	X	X		
g. IEEE Std 379-1972 (ANSI N41.2)	Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems	X	X	X	X	X	X	X	See Reg. Guide 1.53.
h. IEEE Std 384-1974 (ANSI N41.14)	Criteria for Separation of Class IE Equipment and Circuits	X	X	X		X			

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section)							REMARKS
		7.1	7.2	7.3	7.4	7.5	7.6	7.7	
4. Regulatory Guides (RG)									
a. RG 1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	X			X		X		
b. RG 1.7	Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident	X		X		X			
c. RG 1.11	Instrument Lines Penetrating Primary Reactor Containment	X	X	X	X	X	X		
d. RG 1.22	Periodic Testing of Protection System Actuation Functions	X	X	X	X	X	X		
e. RG 1.29	Seismic Design Classification	X	X	X	X	X	X		
f. RG 1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	X	X	X	X	X	X	X	
g. RG 1.32	Use of IEEE Std 303-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"	X			X	X	X		
h. RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	X	X	X	X	X	X		Use in conjunction with Position 3, RG 1.17
i. RG 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	X	X	X	X	X	X		
j. RG 1.62	Manual Initiation of Protection Actions	X	X	X	X		X		
k. RG 1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants	X	X	X	X	X	X	X	
l. RG 1.68	Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors	X	X						

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TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section)							REMARKS
		7.1	7.2	7.3	7.4	7.5	7.6	7.7	
m. RG 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 2.	X	X	X	X	X	X	X	
n. RG 1.75	Physical Independence of Electric Systems	X	X	X		X			
o. RG 1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	X						X	
p. RG 1.89	Qualification of Class IE Equipment for Nuclear Power Plants	X	X	X	X	X	X		
q. RG 1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	X		X					
5. Branch Technical Positions (BTP) EICSB									
a. BTP EICSB 1	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors	X	X	X	X		X		
b. BTP EICSB 3	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System	X			X		X		
c. BTP EICSB 4	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	X			X		X		
d. BTP EICSB 5	Scram Breaker Test Requirements - Technical Specifications	X	X						
e. BTP EICSB 9	Definition and Use of "Channel-Calibration" - Technical Specifications	X	X		X	X	X		
f. BTP EICSB 10	Electrical and Mechanical Equipment Seismic Qualification Program	X	X		X	X	X		
g. BTP EICSB 12	Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service	X	X	X					

TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section)							REMARKS
		7.1	7.2	7.3	7.4	7.5	7.6	7.7	
h. BTP EICSB 13	Design Criteria for Auxiliary Feedwater Systems	X		X					
i. BTP EICSB 14	Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	X	X						
j. BTP EICSB 15	Reactor Coolant Pump Breaker Qualification	X	X						
k. BTP EICSB 16	Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors	X	X						
l. BTP EICSB 18	Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves	X		X	X		X		
m. BTP EICSB 19	Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems	X		X			X		
n. BTP EICSB 20	Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	X		X	X		X		
o. BTP EICSB 21	Guidance for Application of Reg. Guide 1.47	X	X	X	X	X	X		
p. BTP EICSB 22	Guidance for Application of Reg. Guide 1.22	X	X	X	X	X	X		
q. BTP EICSB 23	Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown	X				X			
r. BTP EICSB 24	Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times	X	X	X	X		X		
s. BTP EICSB 25	Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole	X		X	X				
t. BTP EICSB 26	Requirements for Reactor Protection System Anticipatory Trips	X	X						
u. BTP EICSB 27	Design Criteria for Thermal Overland Protection for Motors of Motor-Operated Valves	X		X	X		X		

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CHAPTER 8

ELECTRIC POWER

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8.0 ELECTRIC POWER

8.1 Introduction

The electric power system supplies power for the plant auxiliaries and the overall control and instrumentation system during normal operation, and for the plant protection and safety features during abnormal and accident conditions.

8.2 Offsite Power System

8.2.1 Function

Connection to the offsite transmission grid is provided to supply an off-site source of power to the onsite electric power system. It also provides the means to transmit the unit electric power output to the utility transmission grid.

8.2.2 Description

The basic heavy water reactor plant design, used as a basis for this report, utilizes a single offsite to onsite 138 KV power supply, referred to as the System Service Supply. In the switchyard, a stepdown System Service Transformer provides a 6.9 KV power supply. During normal operation, the auxiliary services supply is shared between the System Service Transformer (SST) and the 6.9 KV Unit Service Transformer (UST), which is the onsite power derived from the station generator output. Either transformer can provide station service load in the event of failure of the other supply.

8.2.3 Design Bases

Unknown.

8.2.4 Safety Evaluation

8.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Offsite Power System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 17 - Electric power systems.
- GDC 18 - Inspection and testing of electric power systems.

8.2.4.2 Assessment

The information available at the time of preparing this report indicated that the plant utilizes a single offsite to on-site power supply. It is realized that a minimum of two independent circuits will be available for plants sited in the USA.

The Quality Assurance Program and record keeping required by GDC 1 and Appendix B of 10 CFR 50 should apply to those portions of the electric power system that are part of the plant.

The transformers and disconnect gear plus the remainder of the electric power system in the plant should be seismically qualified to meet the requirements of GDC 2.

Missile protection should be afforded critical components of the electric power system if they are not redundant or physically separated. This would be required to satisfy the requirements of GDC 4.

8.2.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Seismically qualify essential components of the electric power system.
- GDC 4 - Provide missile protection for essential components of the electric power system.
- GDC 17 - Add a rapid acting disconnect device between the main generator terminals and the taps to the unit service transformer to allow power flow through the generator step-up transformer and the unit station transformer.

On the basis of information available an additional independent connection to the transmission network would be required.

- GDC 18 - Include the electric power system in the inservice inspection program to provide periodic inspection of the system and components to assure operability and ability to function as designed.

8.3 Onsite Power Systems

8.3.1 ac Power Systems

8.3.1.1 Function

Provides the means for distributing ac power to the plant auxiliaries and loads during startup, normal and emergency operation and shutdown.

8.3.1.2 Description

Electric ac power supplies for service power and instrumentation are divided into three categories, based on the service and safety function that must be performed. The definitions are as follows:

Class II - ac loads which must not be interrupted.

Class III - ac loads which can tolerate an interruption, but are required for safety functions.

Class IV - non-safety related ac loads.

The two service transformers (UST and SST) feed six 6.9 KV buses located in the electrical equipment room. Four of these buses supply the Class IV loads, and two supply the Class III loads. All buses have alternate feeds from the service transformers and the Class III buses have connections to the emergency diesel generator sets. The 600 Volt Class IV buses are supplied through a 6.9 KV-600 V transformers from the 6.9 KV Class IV buses. The 600 Volt Class III buses are supplied through similar transformers from the Class III 6.9 KV buses. The transformer impedance is such as to allow starting of the largest 600 Volt motor without excessive voltage drops and to limit the fault level to 40,000 Amps. symmetrical to allow the use of commercially available switchgear.

An automatic transfer system is provided which insures continuity of electrical supply to the Class III and IV 6.9 KV buses in the event of failure

of either the unit or of the system supply. Transfer of the load from the service transformer to the other can be accomplished by two methods.

A parallel transfer is accomplished by closing the incoming supply breaker for the bus and then automatically tripping the alternate supply breaker. The two service transformers will then be paralleled for a few cycles. This system is used for normal transfers, manually initiated after startup and before shutdown. It is also automatically initiated for mechanical trips on the turbine and reactor trips.

An emergency fast open transfer is accomplished by tripping the supply breaker and closing the alternate supply breaker for a bus simultaneously. Because the trip action is faster than the stored energy closing mechanism, there will be a few cycles during which the bus is disconnected from both transformers. The transfer is fast enough so that the voltage and phase difference between the incoming supply and the residual on the motors is small enough to avoid excessive inrush currents. Fast open transfers are initiated by electrical faults on the generator, unit service transformer, main transformer or system service transfer.

The emergency power for the Class III loads is supplied by four diesel generator sets. These are housed in four separate rooms with fire resistant walls. Two units are seismically qualified and the other two (smaller) units are seismically qualified. There are two Class III buses each with two diesel generators connected to it. Each pair of non-seismic units are sized to supply the total shutdown load of the unit. The Class III shutdown loads are duplicated, one complete system being fed from a pair of diesel

generator sets. In the event of failure of Class IV power, the four diesel generators will start automatically. The seismically qualified units only operate a limited number of systems. Canadian General Design Criteria does not superimpose an earthquake coincident with LOCA.

The diesel generator sets will be up to speed, synchronized, and ready to accept load in less than two minutes. Each of the large generators will then automatically energize half of the shutdown load. In this condition, there is no electrical tie between the two buses and they are not required to be synchronized. Each generating unit consists of a salient pole synchronous generator directly coupled to an engine complete with all auxiliaries including lubricating oil system with heat exchangers, engine cooling system with raw service water cooling through a heat exchanger, compressed air starting system, fuel system, excitation system, governor and control. The diesel generator units are rated to meet the maximum power requirements for periods in excess of 72 hours.

Motors of 400 HP and above are supplied from 6.9 KV buses, motors of less than 400 HP are supplied from 600 Volt buses. Motors of less than 100 HP are supplied from combination starter units. Each starter has a molded case circuit breaker and a magnetic contactor. The circuit breaker has instantaneous and in some cases inverse time trip, and has an interrupting capacity of 22,000 Amps symmetrical. The contactor is sized according to NEMA standards, and is provided with three overload relays of the Bimetal or Solder Alloy type sized to protect the motor.

8.3.1.3 Design Bases

Unknown.

8.3.1.4 Safety Evaluation

8.3.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Onsite Power System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 3 - Fire protection.
- GDC 4 - Environmental and missile design bases.
- GDC 5 - Sharing of structures, systems, and components.
- GDC 17 - Electric power systems.
- GDC 18 - Inspection and testing of electric power systems.
- GDC 19 - Control room.
- GDC 35 - Emergency core cooling.
- GDC 37 - Testing of emergency core cooling system.
- GDC 38 - Containment heat removal.
- GDC 40 - Testing of containment heat removal system.
- GDC 41 - Containment atmosphere cleanup.
- GDC 43 - Testing of containment atmosphere cleanup systems.
- GDC 44 - Cooling water.
- GDC 46 - Testing of cooling water system.
- GDC 52 - Capability of containment leakage rate testing.
- GDC 53 - Provisions for containment inspection and testing.

8.3.1.4.2 Assessment

There appear to be significant differences between the Canadian and U.S. approaches to safety related electric equipment. From available information, the Canadian safety classes (Class II and III) are essentially oriented towards reliable system design but do not consider single failures, common mode failures or seismic events. The U.S. safety class (Class 1E) encompasses not only reliable system design including consideration of single failures, common mode failures and seismic events but also includes equipment and component design and fabrication and requires verification of the equipment, components and systems ability to perform their functions under maximum anticipated service requirements and conditions. Based on the above, the onsite ac power system violates the following General Design Criteria:

1	18	41
2	19	43
3	35	44
4	37	46
5	38	52
17	40	53

and associated Regulatory Guides for their implementation.

8.3.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- o Use equipment qualified to meet Class 1E requirements in all Class II and III systems.
- o Install Class II and III equipment in seismic Category I structures.
- o Remove all ties between redundant Class II and III channels.

- o Ensure complete physical and electrical independence and separation is maintained between Class II and III channels throughout.
- o Install three hour rated fire walls between redundant equipment.
- o Redesign the cable spreading area for the control room so that each redundant safety related channel plus the non-safety related channel is enclosed in its own room with three hour rated fire walls.
- o Design emergency diesel generators to be seismically qualified and to handle the total emergency load.
- o Ensure that the diesel-generator units will be up to speed and ready to accept load in 10 seconds or less.
- o Provide diesel-generator units that are rated for an indefinite period of operation.
- o Provide onsite fuel storage for the diesel generators of seven days capacity or of sufficient capacity to operate the units for the time required to replenish the fuel from sources away from the site, whichever is longer.
- o Install cables with insulation systems that will minimize the release of toxic substances when burning.

8.3.2 dc Power Systems

8.3.2.1 Function

Provides the means to supply and distribute dc power to Class I plant auxiliaries.

8.3.2.2 Description

The Class I power supply is comprised of three independent 250 V dc battery banks and three independent 48 V dc battery banks, each equipped with a battery charger fed from Class III buses. This permits complete independence of the supply to the triplicated reactor protective and safety systems. Each battery bank is installed in a separate room.

The 250 V dc network uses molded circuit breakers for distribution protection. The network supplies the power for all the dc motors, and the Class II 600 Volt inverters.

The Class II ac power for instrumentation, computers and critical motor loads is supplied from static inverters which are in turn fed from the 250 V dc Class I batteries. The instrumentation and computers are fed from three 120/208 Volt ac inverters. This ensures complete independence of supply to the triplicated reactor protection instrumentation. The Class II motors are fed from two 600 Volt ac inverters, thus providing segregation between instrumentation and motor loads.

Each inverter is normally supplied through a rectifier battery charger in parallel with a battery. The rectifier receives its supply from a Class III bus.

In the event of a Class III power failure the inverter continues to be supplied from the battery without interruption.

Power for control purposes is supplied from the three 48 V dc batteries and chargers. Distribution and protection for the 48 V dc Class I network is provided by distribution panels using molded case circuit breakers.

8.3.2.3 Design Bases

Unknown.

8.3.2.4 Safety Evaluation

8.3.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of dc Power Systems:

See listing in Section 8.3.1.4.1, above.

8.3.2.4.2 Assessment

Refer to Section 8.3.1.4.2. Additionally, the connection of safety related and non-safety related equipment to safety related power supplies is prohibited unless the non-safety related equipment is qualified as Class 1E equipment per GDC 17 and its implementation to the requirements of Regulatory Guide 1.75.

8.3.2.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- o Install the batteries, chargers, inverters and distribution panels in a seismic Category I structure with three hour fire walls between redundant channels.

- o Qualify all Class I equipment to meet Class 1E requirements.
- o Install a separate non-safety related battery system to supply non-safety related dc power requirements or qualify and separate all dc equipment in the plant per Class 1E requirements.

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9.0 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

The entire flow of fuel through the plant is shown schematically in Figure 9.1-1. The new fuel stored in dry racks in the new fuel storage room of the Service Building is transferred through the equipment lock into the new fuel loading area in the Reactor Building. In the loading area of the reactor building, the fuel bundles are given final visual inspection and are manually loaded into the magazine of the new fuel ports which penetrate into the fueling machine maintenance locks. Refer to Figure 9.1-3. Mechanisms of the ports are motorized and can be controlled remotely.

One of the two fueling machines locks onto the new fuel port where it can accept as many as 10 new fuel bundles into five magazine positions, by loading 2 bundles into each magazine.

9.1.1 New Fuel Storage

9.1.1.1 Function

The purpose of this system is to provide space for storage of new fuel.

9.1.1.2 Description

New fuel is stored in dry racks of the new fuel storage room of the Service Building; refer to Figure 9.1-1.

The new fuel storage room accommodates six month's supply of fuel, in addition to temporary storage of fuel for critical loading.

9.1.1.3 Design Bases

Unknown.

9.1.1.4 Safety Evaluation

9.1.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the New Fuel Storage:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 61 - Fuel storage and handling and radioactivity control.
- GDC 62 - Prevention of criticality in fuel storage and handling.
- GDC 63 - Monitoring fuel and waste storage.

9.1.1.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 are only partially complied with.

The new fuel storage racks should be evaluated to assure they are seismically qualified to meet the intent of GDC 2 for the ground accelerations normally assigned to nuclear power plants sited in the USA. The present construction of the Service Building which houses the new fuel storage is not in accordance with USA requirements of seismic Category I structures, and therefore does not qualify for the storage of fuel.

The Service Building does not meet the requirements of GDC 4 for missile protection, therefore, it is not qualified for storage of fuel.

The new fuel storage does not meet the requirements of GDC 61 due to the design of the Service Building.

GDC 62 is complied with in the new fuel storage design.

The new fuel storage meets the intent of GDC 63.

9.1.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Redesign the Service Building to comply with seismic Category I design requirements.
- GDC 4 - See GDC 2, above.
- GDC 61 - See GDC 2, above.
- GDC 62 - No change.
- GDC 63 - No change.

9.1.2 Spent Fuel Storage

9.1.2.1 Function

The function of this facility is to provide safe storage for spent fuel.

9.1.2.2 Description

Spent fuel storage facilities are provided in the Service Building to accommodate approximately 10 years accumulation of irradiated fuel. A separate compartment is provided to contain spent fuel from leaking fuel bundles. The inside surfaces of the spent fuel bays are finished with an epoxy fiber-glass reinforced liner.

9.1.2.3 Design Bases

Unknown.

9.1.2.4 Safety Evaluation

9.1.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Spent Fuel Storage:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 61 - Fuel storage and handling and radioactivity control.
- GDC 62 - Prevention of criticality in fuel storage and handling.
- GDC 63 - Monitoring fuel and waste storage.

9.1.2.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 are only partially complied with.

The spent fuel storage bays are concrete construction with an epoxy fiberglass reinforced liner. Concrete, as well as the liner, is subject to cracking during an earthquake since it is a brittle material. Cracking would permit leaking of cooling water, and as such, is not an acceptable storage pool design for seismically designed structures as required by GDC 2. In addition, the Service Building design does not meet the intent of a seismic Category I structure.

The Service Building does not meet the requirements of GDC 4 for missile protection, therefore, it is not qualified for storage of fuel.

The spent fuel storage does not meet the requirements of GDC 61 due to the design of the Service Building and the material used for lining the storage pool.

GDC 62 is complied with in the spent fuel storage design.

The spent fuel storage meets the intent of GDC 63.

9.1.2.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Provide a metal liner in the spent fuel storage bay. Redesign the Service Building to comply with seismic Category I design requirements.
- GDC 4 - See GDC 2, above.

GDC 61 - See GDC 2, above.

GDC 62 - No change.

GDC 63 - No change.

9.1.3 Spent Fuel Bay Cooling and Cleanup System

9.1.3.1 Function

The purpose of this system is to remove decay heat from the spent fuel in the spent fuel bay as well as the canned fuel storage bays, where the leaking fuel assemblies are stored. It is also designed to provide clarification of the water for visibility and to maintain the activity of the water at suitably low levels. In particular, the system is designed to:

- a) Maintain the temperature of the storage bay water below 104°F with accumulated fuel over 10 years plus $\frac{1}{2}$ the core placed in the bay over a 20 day period.
- b) Maintain purity and clarity of the water to permit observation of fuel handling operations.
- c) Provide capability to remove suspended and dissolved radionuclides and maintain pool activity to acceptably low levels.
- d) Provide capacity to maintain a minimum water level for shielding during all phases of fuel handling and storage.

9.1.3.2 Description

Decay heat from the spent fuel is removed from the storage bays by circulating the bay water through the spent fuel bay heat exchangers. The Raw Service Water System provides cooling to the Spent Fuel Bay heat exchangers. The system consists of a main storage bay loop from the bay to the pumps and coolers and back to the bay, with a side stream flow through the main storage bay skimmers being directed to pumps and coolers and back to the main bay ion exchanger. The cooling water for the canned fuel storage reception and inspection bays is taken off the system down stream of the

heat exchangers, flows through the bays and together with the flow from the skimmer, flows to the purification pump, ion exchangers and the heat exchanger. By manipulating the line valves, the flow to any part of the system may be varied to suit the purification demand. A vacuum cleaning system is provided for cleaning the bay floors. (Refer to Figure 9.1-2).

All piping that penetrates the fuel bay walls is of sufficient height above the stored fuel to provide adequate shielding in the event of a pipe rupture in the system. Refer to Chapter 15 for a discussion of accidents. Siphoning is prevented by the use of siphon breakers or vents where appropriate.

The receiving bay, canned fuel storage bay and main storage bay may each be emptied for inspection or repair. This process is infrequent and may only be performed when no spent fuel is in the section to be drained. Special portable equipment and piping will be required for this operation.

All bays are lined with a fiberglass epoxy to prevent leakage. Piping, valves and equipment of this system which contact the storage bay water is made of stainless steel or other corrosion resistant material. Components of the system are designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code for Class 3 components.

Two main storage bay circulation pumps are provided. The cooling load during normal operation requires only one circulation pump; however, the second pump may be required should it become necessary to store temporarily partially spent fuel from the reactor core.

A pump and ion exchangers are provided for purification. Normally only one purification pump is expected to operate; however, if the demand is greater

than expected, one pump from the main storage bay system may be used. The ion exchangers will be designed for the dual pump flow condition with one ion exchanger taking 50% of the resulting flow.

Connections to the demineralized water distribution system provide the initial bay fill and make-up, and for transfer flask wash down and spent resin transfer. Water discharged from the bays and leakage from the piping system is collected and discharged to the radioactive liquid waste system. Emergency makeup water is provided from the demineralized water storage tank and also from the plant fire water system.

Instrumentation for the Spent Fuel Bay Cooling and Purification System consists of the following:

- a) Control room indication of the Spent Fuel Bay, the Canned Fuel Bay and the Reception Bay temperature and level with high temperature and high/low radiation level alarm.
- b) Local indication of flow in each of two cooling circuits and the Spent Fuel Bay purification circuit.
- c) Control of the circulating pumps from the control room. Electro Mechanical Indicators are provided on the control room panel to show the status of the pumps.

Control of the storage bays temperature is provided by manually operating valves in the heat exchanger cooling water lines.

The system is located in the basement of the Service Building.

All associated pumps and motor operated valves are provided with Class III emergency power.

9.1.3.3 Design Bases

Refer to Table 9.1-1.

9.1.3.4 Safety Evaluation

9.1.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Spent Fuel Bay Cooling and Cleanup System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 60 - Control of releases of radioactive materials.
- GDC 61 - Fuel storage and handling and radioactivity control.

9.1.3.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 are only partially complied with.

The seismic capability of the spent fuel bay cooling and cleanup system will require an evaluation on the basis of the ground acceleration normally assigned to nuclear power plants sited in the USA to meet intent of GDC 2.

The Service Building does not meet the requirements of GDC 4 for missile protection, therefore, it is not qualified to house the ASME Class 3 components. Provisions for protection of equipment and piping may qualify the system.

The intent of GDC 60 is not met since the epoxy-lined concrete storage bays could crack and leak during an earthquake. Also, insufficient information

was available at the time of this assessment to determine if the Spent Fuel Cooling and Purification System heat exchangers are sized such that the system could perform its intended function in the event of a passive failure.

The intent of GDC 61 will be met when GDC 60 concerns are fulfilled.

9.1.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Redesign the Service Building and evaluate the seismic capability of the spent fuel bay cooling and cleanup system to comply with seismic Category I design requirements.
- GDC 4 - See GDC 2 above.
- GDC 60 - Provide steel liner in the concrete storage bays. Analyze the heat exchangers to assure that the system can perform its intended function in the event of a passive failure.
- GDC 61 - No change if GDC 60, above, requirements are met.

9.1.4 Fuel Handling Systems

9.1.4.1 Function

Fuel handling systems are provided to supply fresh fuel to the core and remove spent fuel from the core in such a manner that shutdown of the reactor is unnecessary.

9.1.4.2 Description

Fuel Changing

The on-power fuel changing equipment is located in the Reactor Building and consists of two identical, unshielded fueling machines, which are operated remotely. The fueling machines are normally stored in two fueling machine maintenance locks and are suspended from tracks. Each set of tracks connects with a bridge at each face of the reactor. Powered shielding doors separate the maintenance locks from the reactor and, when closed, allow access to the fueling machines while the reactor is at full power. Refer to Figures 9.1-3, 9.1-4, and 9.1-5 for configuration of the fuel handling machines and areas.

While in the maintenance locks, the fueling machines can lock on to the new fuel port to accept new fuel, to the service port for maintenance or service, or onto the spent fuel port to discharge spent fuel.

Operations on-reactor are performed under remote control. One of the fueling machines locks on to the new fuel port and accepts up to 10 new fuel bundles, with the other machine accepting 8 for a refueling operation leaving one empty magazine to accept spent fuel. The shielding doors are opened and the two machines travel along the tracks onto the fueling machine bridges at each end of the reactor. The bridges are then raised and the fueling machines are positioned on the bridge so that the machines are located at each end of the selected fuel channel.

A description of the fueling machine is given later in this section.

Both fueling machines move forward and lock on to the fuel channel end fitting to form a secure and leak-tight joint with the channel. Each machine is pressurized to approximately the same pressure that exists in the end fitting and the plugs from the snouts of the machines are removed. The fueling machines remove the fuel channel closures from the pressure tube and store them in a magazine. The machine at the upstream end of the fuel channel removes the shield plug and inserts a new fuel bundle into the channel from a new fuel magazine. The downstream machine removes the shield plug and receives a spent fuel bundle which is also stored in the fuel magazines. Two fuel bundles can be stored in each magazine position. Either machine can load or receive fuel, depending upon the direction of flow of the D_2O in the channel being fueled, fueling normally being with the direction of flow.

When the required number of fuel bundles has been inserted, the shield plugs and channel closures are replaced and the closures checked for leaks. The machines replace the plugs in the snouts and unlock from the end fitting. They are then retracted from the end fitting and returned to the fueling machine maintenance locks, or can be moved to another fuel channel for further fueling operations.

Upon returning to the maintenance lock, the fueling machine containing the spent fuel discharges it through the spent fuel port.

Spent Fuel Handling

The spent fuel handling equipment consists of discharge and transfer equipment in the Reactor Building and spent fuel reception and storage equipment and bays in the Service Building. The transfer of spent fuel between buildings is carried out under water through a transfer canal. Handling and storage space is also provided under water for irradiated parts of the reactivity control mechanisms, which may be removed from the reactor.

The discharge and transfer operations are controlled remotely, while operations in the storage bays are carried out manually under water using long tools, and aided by powered cranes and hoists.

The equipment incorporates devices for canning defective fuel bundles and is arranged to reduce the radiation exposure of personnel when handling defective fuel to acceptable levels.

The discharge equipment comprises two valved spent fuel ports located above the water canal, while the transfer equipment is located in a shielded room and extends down under the water into a reception bay in the Service Building. In operation, the fueling machine seals to the spent fuel port

in the fueling machine maintenance lock and the heavy water level in the head is lowered below the snout. The snout plug is then removed and the machine discharges the spent fuel, two bundles at a time, through the spent fuel ports onto the transfer mechanism. The mechanism lowers the fuel into the light water of the fuel transfer canal and transports it into the reception bay in the Service Building. Defective fuel is manually inserted under water into cans before transfer, to limit the spread of contamination.

The reception bay communicates under water with the main storage bay on one side and with a small defective fuel bay on the other side. In the reception bay, the normal spent fuel is remote manually off-loaded under water from the transporter and placed onto storage trays. Canned defective fuel, however, is routed for storage in the defected fuel bay. In addition to the trays of spent fuel, irradiated elements of the reactivity control mechanisms received in the reception bay are moved into the storage bay for under water storage. Space is also allocated for a shipping cask which would be lowered into the water of the bay for loading the fuel.

Fueling Machines

For descriptive purposes, the fueling machines consists of a head, a suspension and a self-propelled carriage. The machine travels on an elevated track when in the maintenance lock and on the fueling machine bridge when at the face of the reactor. Heavy water, air, electric power and oil hydraulic connections and control signals are supplied to the fueling machine through a flexible catenary which connects the mobile machine to the stationary auxiliary systems.

The function of the fueling machine head is to pick up new fuel from the new fuel port, load it into the reactor, receive spent fuel from the reactor and discharge it into the spent fuel port.

The general arrangement of the fueling machine head is shown in Figure 9.1-6. It is composed of several main assemblies which perform different functions.

The snout assembly is located at the front end of the fueling machine head. It consists of an antenna assembly which ensures that the fueling machine head is in adequate alignment with an end fitting before the head clamps on to it. The snout clamping mechanisms latch the fueling machine head on to an end fitting to form a pressure tight seal. When not attached to a fuel channel or a port, the snout is sealed by a snout plug. When attached to a fuel channel, the snout plug is stored in one of the 12 magazines.

The magazine assembly is located behind the snout assembly and the separators. The magazine housing and end cover are forgings which take the form of a cylindrical pressure vessel with flat heads. The housing and the end cover are fastened together by a two-piece Grayloc clamp and seal ring.

A twelve-position rotatable magazine is located concentrically inside the magazine housing and is driven by an oil hydraulic motor. The twelve magazine positions provide for five fuel stations (holding two bundles each), two channel closures, two shield plugs, one snout plug, one ram adapter and one guide sleeve and tool.

The guide sleeve is a tube with an internal diameter nominally equal to that of the coolant channel. It is inserted into the snout and the end fitting to facilitate the passage of fuel bundles into the magazine.

Two separator assemblies are located between the snout and the magazine assemblies. The separators are used for sensing the passage of fuel bundles

and shield plugs into the machine and for separating the spent fuel bundles from the remainder of the fuel string.

The ram assembly is located immediately behind the magazine assembly. The telescopic rams operate all the plugs, the ram adapter, the guide sleeve and tool, and push fuel bundles into the fuel channel. The rams are enclosed in a cylindrical housing and are driven by two pairs of water-lubricated ball screws arranged within the housing. The ball screws penetrate the pressure boundary at the rear of the housing and are sealed by self-energized hydrostatic face seals. An oil lubricated gearbox and two oil hydraulic motors drive the ball screws.

The head is secured to the carriage through a suspension. This is an assembly which allows the head to align itself with a channel end fitting to reduce the forces exerted on the end fitting by the head. The catenary hoses and cables are connected to the head by quick-disconnect couplings. These, in conjunction with a mechanical disconnect, enable the head to be removed remotely in the event that the spent fuel is stuck in the head and cannot be discharged by normal methods. The same couplings and mechanical disconnect are also utilized for removal of the head for routine maintenance.

The fueling machine carriage is equipped with eight wheels which run on the tracks in the fueling machine maintenance lock and on the fueling machine bridge. It is electrically driven for horizontal traverse, the eight wheels enable it to cross the gaps between the tracks and the bridge. Slow speed is provided for homing motion. The fueling machine bridge is elevated by 4 ball screws along 2 vertical columns.

Auxiliaries and Catenaries

D₂O, air and oil hydraulic systems in addition to electrical power are provided for operation of the fueling machines which are located below the maintenance locks and are accessible at full reactor power. The D₂O system comprises equipment to supply both fueling machines with filtered heavy water at the required temperatures and pressures. D₂O is drawn from and returned to the primary heat transport system storage.

Pressurized air is provided from the plant services, two oil hydraulic power units supply the oil hydraulic devices on the machines.

The operating valves and gauges for all three systems are centralized in a separate set of panels for each machine.

The catenary, connecting the auxiliaries to each fueling machine, consists of a power track and a free loop supported by a trolley.

9.1.4.3 Design Bases

Unknown.

9.1.4.4 Safety Evaluation

9.1.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Fuel Handling System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 14 - Reactor coolant pressure boundary.

- GDC 15 - Reactor coolant system design.
- GDC 50 - Containment design basis.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 52 - Capability for containment leakage rate testing.
- GDC 53 - Provisions for containment testing and inspection.
- GDC 60 - Control of releases of radioactive materials to the environment.
- GDC 61 - Fuel storage and handling and radioactivity control.
- GDC 62 - Prevention of criticality in fuel storage and handling.

9.1.4.4.2 Assessment

The on-line Fuel Handling System is not a safety-related system, but due to the many activities of the system, postulated occurrences could compromise other systems.

The Quality Assurance Program and recordkeeping required by GDC 1 would apply to this system, and the present program only partially complies with the requirements.

The system and its components are qualified to seismic requirements as required by the Canadian Government. A detailed review is required, to satisfy GDC 2, to assess the seismic capability of the crane and hoists used for moving and positioning the fueling machine. The seismic capability of the system would require review for the condition where the fueling machine is attached to a pressure tube fitting during refueling operation.

The system equipment arrangement should be evaluated for the effect of being impacted with a postulated missile generated within the containment and causing the release of radioactive material. This is required to satisfy the requirements of GDC 4.

During refueling operations, the fuel handling machines are connected to the pressurized tubes. An analysis and testing should be performed to assure that this operation could not cause a breach of the reactor coolant pressure boundary. This is required to assure that GDC 14 and 15 are met.

The design of the spent fuel canal requires a detailed evaluation to ascertain if it fully meets the criteria governing the containment design and containment penetration. In particular, GDC 50, 51, 52, and 53 must be met. Sufficient details of the design were not available for this assessment.

The transfer canal and various fuel storage bays should be provided with metal liners in lieu of the present epoxy liner. As designed, the epoxy liner and concrete would be subject to cracking if the plant was subjected to an earthquake and would, thus, provide a potential leak path to the environment. Additional details would be required to fully evaluate the design with regards to compliance with GDC 60, 61, and 62.

9.1.4.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Evaluate design and test the fuel handling machine and bridge for seismic capability. Also, evaluate for potential of damaging the pressure tube assembly when the fueling machine is attached during refueling.

- GDC 3 - Evaluate the pressure tube-fueling machine interfacing for tube rupture during postulated abnormal events.
- GDC 14 - Evaluate the pressure tube-fueling machine interfacing for tube rupture during postulated abnormal events.
- GDC 15 - See GDC 14, above.
- GDC 50 - Evaluate the design and materials of construction of the transfer canal to assure that:
 - o Materials meet fracture toughness requirements.
 - o Containment isolation can be achieved.
- GDC 51 - See GDC 50, above.
- GDC 52 - See GDC 50, above.
- GDC 53 - See GDC 50, above.
- GDC 60 - Provide a liner in the fuel storage bays and transfer canal to mitigate leakage of water from cracked concrete during an earthquake.
- GDC 61 - See GDC 60, above.
- GDC 62 - No change.

TABLE 9.1-1

Spent Fuel Bay Cooling and Purification System Design Data

<u>Item</u>	<u>Data</u>
Main Storage Bay	
Fluid	H ₂ O
Flow Rate	1312 to 2625 gpm
Cooling Capacity	2 MW(th)
Bay Volume	72,000 ft ³
Cleanup Half Life	10 to 20 hours
Discharge and Reception Bay	
Fluid	H ₂ O
Flow Rate	325 to 650 gpm
Cooling Capacity	0.2 MW(th)
Bay Volume	21,000 ft
Cleanup Half Life	3 to 6 hours
Applicable Code	ASME Section III, Class 3
Storage Bay Cooler	
Quantity	1
Type	Tube-in-Shell
Duty	Continuous
Heat Transferred	2 MW(th)
Tube Side Data:	
Fluid	H ₂ O
Flow Rate	1312 to 2625 gpm
Inlet Temperature	95°F

TABLE 9.1-1 (cont'd)

Spent Fuel Bay Cooling and Purification System Design Data

<u>Item</u>	<u>Data</u>
Outlet Temperature	85°F
Design Pressure	90 psig
Design Temperature	120°F
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	950 gpm
Inlet Temperature	65°F
Outlet Temperature	80°F
Design Pressure	150 psig
Design Temperature	200°F
Construction	
Tube Material	Admiralty Brass
Tube Outside Diameter	0.625 in
Shell Material	Carbon Steel
Applicable Codes	
Tube Side	ASME Section III, Class 3
Shell Side	ASME Section VIII
Discharge and Reception Bay Cooler	
Quantity	1
Type	Tube-in-Shell
Duty	Continuous
Capacity	100%

TABLE 9.1-1 (cont'd)

Spent Fuel Bay Cooling and Purification System Design Data

<u>Item</u>	<u>Data</u>
Heat Transferred	0.2 MW(th)
Overall Length	-
Tube Side Data:	
Fluid	H ₂ O
Flow Rate	325 to 650 gpm
Inlet Temperature	95°F
Outlet Temperature	85°F
Design Pressure	90 psig
Design Temperature	120°F
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	95 gpm
Inlet Temperature	65°F
Outlet Temperature	80°F
Design Pressure	150 psig
Design Temperature	200°F
Construction	
Tube Material	Admiralty Brass
Tube Outside Diameter	0.625 in
Shell Material	Carbon Steel
Applicable Codes	
Tube Side	ASME Section III, Class 3
Shell Side	ASME Section VIII

TABLE 9.1-1 (cont'd)

Spent Fuel Bay Cooling and Purification System Design Data

<u>Item</u>	<u>Data</u>
Spent Fuel Bay Cooling and Purification Pumps	
Quantity	3
Type	Centrifugal
Duty	Continuous
Pump Data:	
Fluid	H ₂ O
Flow Rate, each	1312 gpm
Temperature	95°F
Head	90 ft
Design Pressure	90 psig
Design Temperature	120°F
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Power Supply	Class IV
Construction	
Materials in Contact with Fluid	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Spent Fuel Bay Ion Exchange Column	
Quantity	2
Type	Vert. Cylinder, Dished Top, Conical Bottom
Volume	25 ft ³

TABLE 9.1-1 (cont'd)

Spent Fuel Bay Cooling and Purification System Design Data

<u>Item</u>	<u>Data</u>
Resin Capacity	19 ft ³
Type of Screen	Wedge-Wire
Design Conditions	
Maximum Flow Rate, each	325 gpm
Direction of Flow	Downward
Operating Pressure	45 psig
Operating Temperature	95°F
Design Pressure	90 psig
Design Temperature	120°F
Construction	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

FIGURE 9.1-1 FUEL HANDLING SEQUENCE

SPENT FUEL BAY COOLING
AND PURIFICATION SYSTEM

FIGURE 9.1-2

FUEL HANDLING SYSTEM

FIGURE 9.1-3

FUEL HANDLING SYSTEM - SECTION S1

FIGURE 9.1-4

FUEL HANDLING SYSTEM - SECTION S2

FIGURE 9.1-5

FUELLING MACHINE HEAD

FIGURE 9.1-6

9.2 WATER SYSTEMS

Water systems are required to remove heat and provide cooling to plant equipment during normal and emergency operation. The water systems include the following:

- a) condenser circulating water system
- b) raw service water system
- c) recirculated water system

Each system is discussed below.

9.2.1 Condenser Circulating Water System

9.2.1.1 Function

The condenser circulating water system supplies the cooling water requirements of the turbine steam condenser.

9.2.1.2 Description

A typical heavy water reactor plant condenser circulating water system is shown in Figure 9.2-1. The system operates as follows, utilizing once through cooling:

During normal operation, two 50% capacity condenser circulating pumps, located in the pumphouse supply condenser cooling water through filters to a common supply header to the condensers. After passing through the condenser, the cooling water is discharged to the outfall through the discharge conduit with a small flow being recirculated to backwash the traveling screens at the intake structure.

A common practice in Canadian design is to utilize reinforced concrete conduit with an epoxy bond between sections for all piping between the pumphouse and other plant structures.

The condenser circulating water pumps as well as the condenser extraction pumps are supplied by Class IV power buses.

9.2.1.3 Design Bases

Unknown.

9.2.1.4 Safety Evaluation

9.2.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Condenser Circulating Water System:

None - this is not a safety related system.

9.2.1.4.2 Assessment

Once-through-cooling water systems are, in general, not acceptable for plants sited in the USA. The use of water is governed by the Federal Water Pollution Control Act.

9.2.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

Provide cooling towers for the condensers circulating water system.

9.2.2 Raw Service Water System

9.2.2.1 Function

The raw service water system supplies cooling water to the moderator heat exchangers, end shield heat exchangers, spent fuel bay heat exchangers, chillers, emergency core cooling recirculation heat exchangers and the closed recirculated water systems.

9.2.2.2 Description

The raw service water is drawn from the ultimate heat sink at the pumphouse, first passing through trash racks and traveling screens.

Four 33-1/3% capacity double suction axially split casing pumps are mounted vertically in a pit in the pumphouse. The pumps are driven by 1000 HP motors mounted vertically above each pump. Two service water pumps are provided emergency Class III power from one bus and two pumps from the other bus. They are normally driven from the Class IV bus. On failure of the Class IV power supply, only one pump will be driven from the Class III bus to supply essential loads only, and all other loads will be automatically isolated. Refer to Figure 9.2.2 for Class IV and Class III service water requirements.

The pumps discharge into a manifold, and from there into one of two self-cleaning strainers. One of the strainers provides water to the Recirculated Water Heat Exchangers, while the other strainer supplies the station raw water requirements. The strainers remove all particles larger than 250 microns, and are continuously back washed. They are provided with isolation and by-pass valves. Refer to Figure 9.2-3 for a schematic flow diagram.

Two types of piping are used for the Raw Service Water System, reinforced concrete pipe for underground service and ductile iron for service indoors. Joints between concrete pipe sections are sealed with elastomeric joints, and iron joints are either flanged, or use mechanical sleeve type couplings. Elbows and tees are also ductile iron.

9.2.2.3 Design Bases

Unknown.

9.2.2.4 Safety Evaluation

9.2.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Raw Service Water System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 44 - Cooling water.
- GDC 45 - Inspection of cooling water system.
- GDC 46 - Testing of cooling water system.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 52 - Capability for containment leakage rate testing.
- GDC 53 - Provisions for containment testing and inspection.
- GDC 54 - Piping systems penetrating containment.
- GDC 57 - Closed system isolation valves.

9.2.2.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 are only partially complied with. All components should be in accordance with

ASME III, Class requirements, and Class 2 if they penetrate containment.

Use of concrete pipe and ductile iron pipe would not qualify for seismic Category I piping systems or meet the requirements of GDC 2.

The cooling water system does not meet the single failure criteria for passive components, therefore, it does not comply with GDC 44.

The cooling water system will require an inservice inspection program to assure leak-tightness of the system and functioning of the pumps and valves as required by GDC 45 and 46.

Cooling water supply and return lines that penetrate containment must comply with GDC 51, 52, 53, 54, and 57, and include:

- o Pipe material selected to have the required fracture-toughness at operating and/or design temperature.
- o Provisions for leak-testing and leak-rate testing.
- o Piping and penetration included in the inservice inspection program.
- o Provide a containment isolation valve located on the outside of containment.

9.2.2.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- The system and components should be designed to ASME III, Class 3 requirements. Piping and valves penetrating containment should meet Class 2 requirements.

- GDC 2 - Change material of piping from concrete and ductile iron to carbon steel or concrete-lined carbon steel pipe.
- GDC 44 - Provide redundant pipe headers to meet single failure criteria.
- GDC 45 - Provide an inservice inspection program to assure leak-tight-
GDC 46 ness of the system and functioning of the pumps and valves.
- GDC 51 - Piping penetrating containment should have proper fracture-toughness for the operating and/or service temperature.
- GDC 52 - Include piping that penetrates containment and the contain-
GDC 53 ment isolation valves in the inservice inspection program.
- GDC 54 - Provide containment isolation valve for each line outside of containment.
- GDC 57 - See GDC 54, above.

9.2.3 Recirculated Water System

9.2.3.1 Function

The recirculated water system is a closed loop cooling system utilizing demineralized water to items of equipment as shown in Figure 9.2.2 for normal and emergency operation.

9.2.3.2 Description

The recirculated water is cooled by four 33-1/3% heat exchangers and pumped by four 33-1/3% end suction pumps, all located in a bay of the pumphouse. Isolation valves are provided for each item of equipment to permit maintenance without shutting down the system. Refer to Figure 9.2-3 for a flow diagram.

Depending on conditions, the shells of the heat exchangers will be carbon steel, the tubes aluminum brass, with the Raw Service Water passing through the tube side of the heat exchanger. The pumphouse is arranged to allow the withdrawal of the tube bundles from the shells.

The pumps are double volute type with pullout construction. Pump casings are cast iron, the impellers are bronze.

In the event of a Class IV power failure, the water supply to all equipment that is no longer essential is automatically shut off and one pump will provide cooling water for essential equipment.

All pumps have their power supply from Class III power - 2 from one bus and 2 from the other, with relays arranged to permit only one pump to run on loss of Class IV power.

A header tank is provided in the system to compensate for coolant volume variations. All pipe and equipment is of carbon steel material.

9.2.3.3 Design Bases

Unknown.

9.2.3.4 Safety Evaluation

9.2.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Recirculated Water System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 44 - Cooling water.
- GDC 45 - Inspection of cooling water system.
- GDC 46 - Testing of cooling water system.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 52 - Capability for containment leakage rate testing.
- GDC 53 - Provisions for containment testing and inspection.
- GDC 54 - Piping systems penetrating containment.
- GDC 57 - Closed system isolation valves.

9.2.3.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 are only partially complied with all components should be in accordance with ASME III, Class 3 requirements and Class 2 if they penetrate containment.

The system appears to meet the requirements of GDC 2, but would require verification that the seismic loads are included in the analysis.

The cooling water system does not meet the single failure criteria for passive components, therefore, it does not comply with GDC 44.

The cooling water system will require an inservice inspection program to assure leak-tightness of the system and functioning of the pumps and valves as required by GDC 45 and 46.

Cooling water supply and return lines that penetrate containment must comply with GDC 51, 52, 53, 54, and 57, and include:

- o Pipe material selected to have the required fracture-toughness at operating and/or design temperature.
- o Provisions for leak-testing and leak-rate testing.
- o Piping and penetrations included in the inservice inspection program.
- o Provide a containment isolation valve located on the outside of containment.

9.2.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
The system and components should be designed to ASME III, Class 3 requirements. Piping and valves penetrating containment should meet Class 2 requirements.
- GDC 2 - No change.
- GDC 44 - Provide redundant pipe headers to meet single-failure criteria.

- GDC 45 - Provide an inservice inspection program to assure leak-tight-
- GDC 46 ness of the system and functioning of the pump and valves.
- GDC 51 - Piping penetrating containment should have proper fracture-
- toughness for the operating and/or service temperature.
- GDC 52 - Include piping that penetrates containment and the contain-
- GDC 53 ment isolation valves in the inservice inspection program.
- GDC 54 - Provide containment isolation valve for each line of
- containment.
- GDC 57 - See GDC 54, above.

9.2.4 Ultimate Heat Sink

9.2.4.1 Function

The purpose of the ultimate heat sink is to provide a source of cooling water for waste heat rejection during all modes of operation, including anticipated operational occurrences and environmental phenomena.

9.2.4.2 Description

The PHWR design reviewed incorporates a large body of water such as a lake or river as the ultimate heat sink. The pumphouse and associated piping systems are not designed against tornadoes or earthquake.

9.2.4.3 Design Bases

Does not apply.

9.2.4.4 Safety Evaluation

9.2.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Ultimate Heat Sink:

None as presently designed.

9.2.4.4.2 Assessment

The Federal Water Pollution Control Act does not permit once-through-cooling, in general. The design of the plant reviewed for this assessment does not meet the intent of the Act since it incorporates a large body of water as the ultimate heat sink.

9.2.4.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

Provide redundant seismic Category I ultimate heat sink cooling towers to supply an emergency water supply to essential systems and components.

CONDENSER CIRCULATING WATER SYSTEM
FIGURE 9.2-1

CLASS IV POWER
SERVICE WATER REQUIREMENTS

CLASS III POWER
SERVICE WATER REQUIREMENTS

RAW SERVICE WATER AND RECIRCULATED
WATER SYSTEM HEAT LOADS
FIGURE 9.2-2

**SERVICE WATER AND RECIRCULATED WATER SYSTEM
FLOW DIAGRAM**

FIGURE 9.2-3

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

9.3.1.1 Function

The compressed air system provides compressed air for general services and protective mask and suits in addition to instrumentation requirements.

9.3.1.2 Description

The compressed air system consists of three air compressor, cooler, and separator trains all feeding into a common header through which two receivers are fed. Air for general services and protective mask and suit stations comes directly from the receivers. Instrument air first passes through a regenerative dryer and is then filtered prior to distribution. Refer to Figure 9.3-1 for a diagram of the system.

9.3.1.3 Design Bases

Refer to Table 9.3-1.

9.3.1.4 Safety Evaluation

9.3.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Compressed Air System:

None, as presently designed.

9.3.1.4.2 Assessment

To meet U.S. criteria, the air receiver and piping systems to the points of application having a safety-related function must meet the single-failure criteria. The system, as designed, does not comply with this requirement.

9.3.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- o Redesign to comply with single failure criteria.
- o Design to seismic Category I, ASME III, Class 3 requirements, as applicable.
- o Provide a Quality Assurance Program.
- o Provide an inservice inspection program.

9.3.2 Process Sampling System

9.3.2.1 Function

The purpose of the Process Sampling System is to allow samples to be taken from the plant process auxiliaries to ensure that the auxiliary systems are operating within design limits.

9.3.2.2 Description

The various sampling systems are discussed with the system descriptions in Chapters 4.0 and 5.0.

9.3.2.3 Design Bases

Unknown.

9.3.2.4 Safety Evaluation

9.3.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Process Sampling System:

See Sections 5.4.6 and 5.4.11.

9.3.2.4.2 Assessment

See Sections 5.4.6 and 5.4.11.

9.3.2.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable criteria:

None.

9.3.3 Equipment and Floor Drainage System

9.3.3.1 Function

The function of this system is to collect liquids that may have leaked from the various plant auxiliary systems.

9.3.3.2 Description

The drainage system is sub-divided as follows:

1. Inactive Drainage
2. Active Drainage
3. Sanitary Drainage

(A) Inactive Drainage

This drainage system collects most of the inactive effluents (pump-house, turbine building) and discharges them directly into the ocean.

(B) Radioactive Drainage

This drainage system is also sub-divided in relation to the degree of contamination of the collected effluents, either low activity or normal activity drainage.

The low activity drainage system collects most effluents from the personnel showers, change rooms, the laundry and the non-active laboratories and floor drains.

These effluents are normally discharged into the ocean without treatment to remove activity. However, in-line monitors are provided to detect any abnormal contamination and to direct the effluents to the liquid waste management system.

The normal activity waste system collects most effluents from the reactor building, such as excess water from ion exchange resin slurring, floor drain; as well as the decontamination center, D₂O clean-up and upgrading, active laboratories and floor drains from the Service Building.

These effluents are discharged directly into the liquid waste management system holdup tanks.

Floor drains located in normally accessible areas of the Reactor Building are connected in the basement to a sump which automatically spills into the radioactive waste system.

The inactive, low active and normal active wastes from floor drains are discharged by gravity, either directly to the ocean or into the waste treatment system. However, at times, they may be drained into sumps provided for this purpose. They are then pumped to the ocean or to the liquid waste treatment system. These sumps are mainly in the turbine building in places where the floor level is low.

9.3.3.3 Design Bases

Unknown.

9.3.3.4 Safety Evaluation

9.3.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Equipment and Floor Drainage System:

GDC 1 - Quality standards and records.

GDC 2 - Design bases for protection against natural phenomena.

GDC 60 - Control of releases of radioactive materials to the environment.

GDC 64 - Monitoring radioactivity releases.

9.3.3.4.2 Assessment

The drainage system does not comply with GDC 60 and 64.

9.3.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

Redesign the system to meet the effluent discharge limits of 10 CFR Part 20.

9.3.4 Chemical and Volume Control System

Refer to the discussion on Moderator Liquid Poison System, PHTS Purification System and Moderator Purification System in Chapters 4.0 and 5.0.

FIGURE 9.3-1 COMPRESSED AIR SYSTEM

Table 9.3-1

Compressed Air System Design Data

<u>Item</u>	<u>Data</u>
Compressor:	
Number	3
Type	Single Stage, double acting
Capacity	400 cfm
Discharge Pressure	115 psig
Receivers:	
Number	2
Volume	200 ft ³
Instrument Air Dryer	
Number	1
Type	Dry Desiccant
Capacity	400 cfm

9.4 Air Conditioning, Heating, Cooling and Ventilation Systems

9.4.1 Function

The purpose of this auxiliary system is to provide heating and cooling to all buildings in the plant during all modes of operation, including operational occurrences.

9.4.2 Description

The Service, Turbine, Water Treatment and Administrative Buildings are heated by hot water produced by a steam-water heat exchanger, supplied with steam produced by an oil heated stand-by boiler. The steam from this boiler is also used for the heavy water upgrading system.

(A) Reactor Building

The internal rooms of the Reactor Building pick up heat from the boiler, pumps and equipment when the reactor is running, and is greater than the building losses.

When the reactor is shut down in the winter, heating coils in the ventilation ducting, in conjunction with portable heaters, serve to keep the building warm.

Local fan-coil cooling units, supplied with cooling water from the service water system are used to cool the Reactor Building.

The normally accessible areas are ventilated by a once-through flow of filtered outdoor air. The boiler and fueling machine rooms are not normally ventilated but these rooms may be purged with ventilation air, if required. Normally, air within the boiler and fueling machine rooms is recirculated in a separate system to recover heavy water vapor and restrict tritium levels in the air.

The building atmosphere is maintained negative to its surroundings and the general direction of air flow within the building is from areas of low to high contamination probability.

Air leakage into the boiler and fueling machine rooms is minimized by ventilation barriers, and a filtered exhaust flow, equal to the air leakage rate, is provided to prevent flow reversals from the normally inaccessible to the accessible areas of the building.

Ventilation exhaust air discharged from the building is continuously monitored for activity and filtered, if necessary, before release to the plant discharge air duct. Containment isolation is by two valves in series in each of the ventilation duct penetrations in the containment wall. The isolation valves are automatically closed in the event of a loss-of-coolant accident. After such an accident any residual pressure may be released to the discharge duct via an emergency filtering system designed to maintain subatmospheric pressures within the building.

(B) Service Building

A one-way air conditioning system is installed in the Service Building without any air recycling to provide a suitable temperature and humidity in the various rooms by continuous treatment of fresh air.

The fresh treated air follows a forced irreversible flow from zones with a low radioactive contamination to areas having a higher radioactive contamination, without the possibility of any air recycling.

Air is drawn into the building, filtered, and heated or cooled as required. Before being discharged from the building, the air is filtered and monitored for activity, with the degree of filtration depending on the degree of contamination in the air.

The hot water heating system with heating coils in the ventilation inlet ducts is designed to maintain an indoor temperature of 73°F in winter when the ventilation system is running.

An air conditioning system for the Control Room is provided to ensure suitable temperatures and humidity in the various rooms as well as an adequate renewal of ventilated fresh air.

(C) Turbine Building

A cooling system is provided in the Turbine Building which, in conjunction with heaters, will maintain the building temperature between 65°F and 90°F with a maximum temperature of 105°F in the areas where considerable heat rises occur. These include the condenser area, the feed-water heater area and the deaerator area. Cooling is provided from coils and fans with cooling water supplied from the recirculated water system.

In the winter time, ventilation air is drawn from outside the building, mixed with recirculated air inside the building and circulated as required. Ventilators are provided at the hot pieces of equipment to expel the hot air, and special provisions are provided to ventilate the areas where noxious gases may accumulate, including the battery room and the chemical storage areas.

9.4.3 Design Bases

Unknown.

9.4.4 Safety Evaluation

9.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Air Conditioning, Heating, Cooling, and Ventilation Systems.

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 19 - Control room.
- GDC 50 - Containment design basis.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 52 - Capability for containment leakage rate testing.
- GDC 53 - Provisions for containment inspection and testing.
- GDC 54 - Systems penetrating containment.
- GDC 56 - Primary containment isolation.
- GDC 60 - Control of releases of radioactive materials to the environment.
- GDC 64 - Monitoring radioactivity releases.

9.4.4.2 Assessment

The ventilation system within and penetrating the containment is a non-safety related system. Those portions of the system located within the containment that could impact on safety-related systems, if torn from their supports during an earthquake, should be designed to seismic Category I requirements. Those portions of the system that penetrate containment should

be designed to seismic Category I requirement, meet Class 2 requirements, and be provided with isolation valves both inside and outside of containment. This requires a complete redesign and analysis of the system.

The air-conditioning system for the Control Room should be designed to seismic Category I requirements, meet Class 3 design requirement, and provide air handling system with redundant seismic Category I remote air intakes. The design of the air handling system should include provisions for mitigating the effects of toxic chemical releases outside the Control Room.

9.4.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Provide seismic restraints for the air handling system located in the containment and service building to prevent the parts from tearing loose during an earthquake.
- GDC 19 - Design the HVAC system to seismic Category I requirements and provide redundant seismic Category I remote air intakes.
- GDC 50 - The portion of the ventilation system that penetrates containment and its isolation valves should be designed to seismic Category I requirements and to Class 2 requirements.
- GDC 51 - Materials used in that portion of the ventilation system that penetrate containment and its isolation valves shall be selected for its fracture toughness properties.

- GDC 52 - The design of the containment ventilation system, including its containment isolation valves, shall not preclude performing a containment leak rate test.
- GDC 53 - Portions of the ventilation system that form the containment boundary and containment isolation system should be included in the inservice inspection program for the containment.
- GDC 54 - Those portions of the containment ventilation system that penetrate containment should be provided with containment isolation valves both inside and outside of containment.
- GDC 56 - See GDC 54, above.
- GDC 60 - No change providing the containment isolation valves of the ventilation system are certified for leaktightness and that the control system will automatically actuate the valves when excess radioactivity is detected in the containment.
- GDC 64 - No change.

9.5 Fire Protection System

9.5.1 Function

The purpose of the Fire Protection System is to provide the station with on-site firefighting equipment and systems.

9.5.2 Description

(A) Reactor Building

Fire protection within the Reactor Building is by portable fire extinguishers and by automatic operation of ventilation fire dampers. Portable extinguishing equipment is supplied at the entrance to the normally inaccessible areas to provide fire protection while work or maintenance is taking place in these areas.

(B) Turbine and Service Buildings

A fire water system is provided for outdoor hydrants and indoor dry-pipe deluge systems, hose cabinets and manually-operated sprinkler systems.

Protection of the exterior areas of all buildings, and protection of equipment in the switchyard is provided by the hydrant system. Protection of the main and service transformers is by means of dry-pipe deluge systems.

The turbine generator diesel and oil storage areas are normally protected by manually-operated sprinklers of the fog-nozzle type. Smoke detectors are provided in conjunction with the above system to initiate alarms.

Less hazardous areas inside the Turbine and Service Buildings are protected by fire hose cabinets supplemented by portable dry chemical extinguishers.

Hose cabinets are supplied to enable any normally accessible area of the Service and Turbine Buildings to be reached with two water streams. In areas where water streams may inadvertently come in contact with live electrical equipment, hose nozzles capable of producing a water fog are supplied.

9.5.3 Design Bases

Refer to Table 9.5.1.

9.5.4 Safety Evaluation

9.5.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Fire Protection System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 3 - Fire protection.

9.5.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with.

From the information available it appears that flammable materials, including power wire insulation, are specified in the construction of the plant. It also appears that a primary fire protection water reservoir has not been provided.

9.5.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - Evaluate those portions of the fire protection system, that are located in seismic Category I structures, for their seismic capabilities.
- GDC 3 - Minimize the use of flammable materials in the construction of the plant. In particular, minimize the use of flammable insulation on power wiring.

Provide a fire protection system water reservoir.

TABLE 9.5-1

Fire Protection System Design Data

<u>Item</u>	<u>Data</u>
Fluid	Fresh Water
Number of Pumps	2 x 100% and Pressurizing Pump
Type	Horizontal
Total Flow (main/pressurizer)	1440 gpm/60 gpm
Pump Discharge Pressure	75 psig
Motor Rating (main/pressurizer)	40 BHP/7.5 BHP

9.6 Diesel Generator Auxiliary Systems

9.6.1 Fuel Storage & Transfer System

9.6.1.1 Function

The purpose of this system is to provide fuel oil storage and transfer capability to assure a supply of fuel oil for the operation of one emergency diesel generator for a minimum of 7 days at the maximum diesel loading without refueling.

9.6.1.2 Description

Each diesel generator unit is supplied from a fuel oil supply system which consists of an underground storage tank and a piping system with associated pumps, filters and controls. Also, each diesel has an indoor day tank with capacity for 8 hours operation. Two transfer pumps are provided between the storage tank and day tank, such that an active failure of one pump will not preclude operation of the system.

The fuel is gravity fed from the day tank to the engine.

9.6.1.3 Design Bases

Unknown.

9.6.1.4 Safety Evaluation

9.6.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of Fuel Storage & Transfer System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 5 - Sharing of structures, systems, and components.

GDC 17 - Electric power systems.

9.6.1.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 are only partially complied with. The system and components should meet requirements of ASME III, Class 3.

The system should be evaluated to assure that the components are seismically qualified to meet the intent of GDC 2 for the ground accelerations normally assigned to nuclear power plants sited in the USA.

The present design meets the intent of GDC 5 since each diesel engine has its own separate fuel oil storage tank.

To assure the availability of the diesel-generators, the fuel storage and transfer system should be designed to meet the intent of ASME III, Class 3 and thus meet the requirements of GDC 17.

9.6.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50. The system and components should meet the intent of ASME III, Class 3 requirements.

GDC 2 - No change recommended pending review of seismic analysis.

GDC 5 - No change.

GDC 17 - See GDC 1, above.

9.6.2 Diesel Generator Cooling System

9.6.2.1 Function

The purpose of this system is to provide necessary cooling for the diesel generator unit during its operation and to maintain the engine at an acceptable starting temperature under standby conditions. Also, the system should provide cooling for the lube oil via the lube oil cooler.

9.6.2.2 Description

Cooling is provided by the Recirculated Water Systems (RWS), and the associated RWS pumps are provided with Class III emergency power.

9.6.2.3 Design Bases

Unknown.

9.6.2.4 Safety Evaluation

9.6.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Diesel Generator Cooling System:

GDC 1 - Quality standards and records.

GDC 2 - Design bases for protection against natural phenomena.

GDC 17 - Electric power systems.

9.6.2.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 are only partially complied with. The system and components should meet requirements of ASME III, Class 3.

The system should be evaluated to assure that the components are seismically qualified to meet the intent of GDC 2 for the ground accelerations normally assigned to nuclear power plants sited in the USA.

A failure of the diesel generator cooling system heat exchanger or the recirculated water supply would result in engine failure through overheating. Lack of provision for a separate cooling water system of 100-percent capacity for each diesel engine does not meet the requirements of single failure criteria for independence. The system should meet the requirements for seismic Category I and ASME III, Class 3.

9.6.2.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
The system and components should meet the intent of ASME III, Class 3 requirements.
- GDC 2 - No change recommended pending review of seismic analysis.
- GDC 17 - Provide each diesel engine with its independent, 100-percent capacity, cooling water system. The system and components should meet the intent of requirements for seismic Category I and ASME III, Class 3.

9.6.3 Diesel Generator Starting System

9.6.3.1 Function

The purpose of this system is to provide automatic starting power to the diesel generator.

9.6.3.2 Description

The diesel generators are air-started. Two air reservoirs are provided for each diesel. Each reservoir has two air compressors, one driven by an electrical motor and the other by an internal combustion engine to provide complete black starting capability of the diesel. The system is designed to start the diesel within 30 seconds.

9.6.3.3 Design Bases

Unknown.

9.6.3.4 Safety Evaluation

9.6.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Diesel Generator Starting System:

GDC 1 - Quality standards and records.

GDC 2 - Design bases for protection against natural phenomena.

GDC 17 - Electric power systems.

9.6.3.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 are only partially complied with. The system and components should meet requirements of ASME III, Class 3.

The system should be evaluated to assure that the components are seismically qualified to meet the intent of GDC 2 for the ground accelerations normally assigned to nuclear power plants sited in the USA.

The single failure criteria is met since each diesel engine is provided with two independent starting systems. The capacity of the air storage tanks was not known at the time of this study, hence, the number of starting cycles could not be evaluated.

9.6.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.

The system and components should meet the intent of ASME III, Class 3 requirements.

GDC 2 - No change recommended pending review of seismic analysis.

GDC 17 - No change recommended pending review of air quantity requirements per engine start and the capacity of the air storage tank. The system and components should meet the intent of requirements for seismic Category I and ASME III, Class 3.

9.7 HEAVY WATER MANAGEMENT

The total inventory of heavy water required for the operation of one reactor unit is 1,024,030 lbs. An itemized list of D₂O inventories associated with heavy water management systems is given in Table 9.7-1.

9.7.1 D₂O Supply System

9.7.1.1 Function

The D₂O Supply System is designed to receive and store D₂O from drums and tank trucks and to pump it to the moderator or heat transport system during initial filling. In addition, the four storage tanks are capable of containing the inventory of either the moderator system or the heat transport system in the event that draining for maintenance is required. The tanks can also store high isotopic D₂O during normal reactor operation.

9.7.1.2 Description

The D₂O Supply System flow diagram is shown in Figure 9.7-1. Four D₂O storage tanks with a capacity of 2500 ft³ each and two pumps are provided. This equipment and the valves and piping associated are located in the service building. Either one or both pumps may be used to transfer D₂O to the moderator or heat transport systems. One of the pumps is used to establish recirculation flow before a sample is taken through the hypodermic sample station and to supply D₂O to the moderator deuteration-deuteration system.

Fresh D₂O is added to the storage tanks through the D₂O addition station from drums or from tank trucks by connecting a flexible hose to the tank inlet header. D₂O from the calandria can be transferred to the storage tanks by gravity or by using the moderator pumps. Similarly, the transfer

of D₂O from the heat transport storage tank is made by gravity. It is important to prevent mixing of the moderator and heat transport heavy water because of the difference in tritium concentration.

The D₂O supply system is classified as ASME Section III, Class 3.

9.7.1.3 Design Bases

Refer to Table 9.7-2.

9.7.1.4 Safety Evaluation

9.7.1.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the D₂O Supply System:

None.

9.7.1.4.2 Assessment

This system has no safety function to perform.

9.7.1.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

None.

9.7.2 D₂O Vapor Recovery System

9.7.2.1 Function

Three separate vapor recovery systems are provided in each Reactor Building to maintain a dry atmosphere in areas which are subject to D₂O leakage during operation or shut-down maintenance. The requirements of these systems are:

- (a) The recovery of D₂O leakage within the dried areas at varying escape rates.
- (b) The reduction of tritium activity within the dried areas to minimize the contamination spread and internal dose rates to station personnel in adjacent accessible areas.
- (c) The maintenance of D₂O areas requiring frequent personnel entry as non-suit areas under normal operating circumstances to reduce maintenance times and exposure to gamma fields in the area to a minimum.
- (d) Minimize downgrading caused by air or light-water leakage into D₂O areas.
- (e) The reduction of D₂O losses and activity release rates due to purging of D₂O areas during shut-down or routine maintenance.

9.7.2.2 Description

The three separate vapor recovery systems which consist of dryers, ductwork, D₂O collection tanks, transfer pumps and piping are located in the Service Building at grade elevation.

Recovery of adsorbed water vapor occurs in the reactivation condensers located above the dryer vessels. The resultant water is pumped to one of

a series of collection tanks located in the D₂O management - upgrading area. Recovered D₂O is segregated according to degree of downgrading and tritium activity.

Reactor building areas are segregated with three separate D₂O vapor recovery systems as follows (see Figure 9.7-2):

- (a) Major areas subject to Heat Transport System D₂O leakage which are accessible only during reactor shut-down. Included in this system are:
 - (i) Boiler room, including shut-down cooler area.
 - (ii) Fueling machine operating areas.
 - (iii) Moderator room (exclusive of enclosure around equipment).

Design conditions for this system are 0°F dew-point at a total leakage rate for all areas of 31 lb/hr (D₂O + H₂O). Three drier units are provided for this system each rated for 4000 scfm of a adsorption air flow, with a separate 500 scfm rated pressure balance exhaust drier to maintain conditions as outlined above. A continuous negative pressure balance of the dried area must be maintained with respect to the adjacent accessible areas to prevent tritium from spreading to the latter areas.

- (b) Areas requiring frequent access including:
 - (i) Fueling machine maintenance locks.
 - (ii) Fueling machine auxiliary equipment rooms.
 - (iii) PHTS and moderator instrumentation rooms.
 - (iv) Monitoring rooms.

One 4000 scfm rated drier unit is provided for these areas. It is designed to maintain a room dew-point of -30°F at a total recovery rate of 2.2 lb/hr of 10% isotopic D_2O .

(c) Area inside the enclosure around moderator equipment

One 2000 scfm rated drier unit is provided for this area with a capability to recover 5.5 lb/hr of heavy water at 0°F dew-point. The moderator equipment enclosure is maintained at a negative pressure balance with respect to the areas included in system (a) or (b) as D_2O leakage from the moderator system equipment has a higher tritium activity than D_2O collected in the other two areas.

The driers, heat exchanger housing, air-handling equipment are all designed to full containment pressure according to the requirements for ASME Section III Class MC. Ductwork and dampers located outside containment are according to ASME Section III Class 2 for the same design conditions.

Dew point and temperature measurements are provided at the inlet and outlet of all driers. The driers are operated from local field control panels supplied with each drier. Remote status indicators and annunciation are provided in the control room for each drier. Drier malfunction annunciation is provided both locally and in the control room.

Indication of the D_2O Collection Tank Level and high level alarm are also provided in the control room.

Since the three D_2O vapor recovery systems are extensions of the containment envelope, two pneumatically-operated dampers in series are used to isolate each system on high pressure or high activity in the containment.

The drain valves of the D₂O collection tanks from the boundary of the extended containment system. They are two pneumatically-operated drain valves in series which are normally closed. During draining of the tanks, a high pressure or high activity signal automatically closes the valves.

The negative pressure in the Reactor Building is maintained by extracting air during the vapor recovery cycle. The air is double-dried before being discharged to the action exhaust system. Control of the net exhaust is made with a damper controlled from a manual control station in the control room.

9.7.2.3 Design Basis

Refer to Table 9.7-3.

9.7.2.4 Safety Evaluation

9.7.2.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the D₂O Vapor Recovery System:

- GDC 1 - Quality standards and records.
- GDC 12 - Design bases for protection against natural phenomena.
- GDC 50 - Containment design basis.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 52 - Capability for containment leakage rate testing.
- GDC 53 - Provisions for containment inspection and testing.
- GDC 56 - Primary containment isolation.

9.7.2.4.2 Assessment

The D₂O Vapor Recovery System performs no safety function but those portion

of the system that penetrates containment should comply with containment isolation requirements, and those portions of the system located in the containment shall be seismically qualified.

In addition, the Quality Assurance Program for those applicable portions of the system as required by GDC 1 and Appendix B of 10 CFR 50 are only partially complied with.

9.7.2.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - No change pending review of the seismic analysis.
- GDC 50 - Selected materials that meet fracture toughness requirements for those portions of the system that penetrates containment. Provide capability of the penetrations to permit leak testing and rearrange containment isolation valves with one inside and the other one outside of containment.
- GDC 51 - See GDC 50, above.
- GDC 52 - See GDC 50, above.
- GDC 53 - See GDC 50, above.
- GDC 56 - See GDC 56, above.

9.7.3 D₂O Upgrading System

9.7.3.1 Function

The D₂O upgrading system is designed to separate solutions of D₂O and of H₂O by distillation into a discard stream of two percent D₂O and a product stream of 99.8 percent D₂O. The feed stream isotopic varies from greater than two percent to less than 99.8 percent D₂O.

9.7.3.2 Description

A flow diagram of the D₂O upgrading system is shown in Figure 9.7-4. The distillation column contains a copper gauze packing which promotes intimate contact between the falling liquid and rising vapor. The reboiler generates vapor at the bottom of the tower and the head condenser refluxes the vapor.

The upgrading tower is operated at a pressure of 120 mm Hg to take advantage of the increased relative volatility of H₂O and D₂O. The tower consists of a number of nearly identical flanged sections, a flanged sump mounted on a support skirt. Each section contains a packing of copper gauze, a packing support, a reflux distributor and a reflux collector. Each of these sections also has two sight glasses, a manhole and a sample valve.

Only one of the several feed points provided is used at one time. The D₂O concentration of the feed determines which feed point is used. The circulating pumps transfer liquid from the sump to the reboiler.

Infrared analyzers are used to monitor the isotopic concentration of the bottom product take-off flow and head discard take-off flow. A cold trap is provided to recover D₂O which might otherwise be lost through the vacuum unit.

To minimize the mixing of low tritium D₂O from the heat transport system and high tritium D₂O from the moderator system, it is necessary to drain the distillation tower one or two times per year to separately upgrade D₂O from these two sources.

9.7.3.3 Design Basis

Refer to Table 9.7-4.

9.7.3.4 Safety Evaluation

9.7.3.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the D₂O Upgrading System:

None.

9.7.3.4.2 Assessment

The D₂O Upgrading System performs no safety function.

9.7.3.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

None.

9.7.4 D₂O Cleanup System

9.7.4.1 Function

The D₂O cleanup system removes dissolved, particulate, and organic impurities from recovered D₂O and produces a product suitable for processing in the D₂O Upgrading System.

9.7.4.2 Description

A flow diagram of the D₂O cleanup system is shown in Figure 9.7-3. The water to be cleaned is pumped from the downgraded D₂O storage tanks through the D₂O cleanup system to the product tanks. The system is provided with one carbon filter which removes particulates and organic material and three ion exchange columns in series which remove dissolved ionic impurities.

The charcoal and ion exchange resin are added to the vessels manually. Removal of spent resin is by slurring with high pressure demineralized water. Shielding is provided around the charcoal filter and ion exchange columns. The cleanup system is fabricated of stainless steel and is designed to ASME Section III Class 3.

9.7.4.3 Design Basis

Refer to Table 9.7-4.

9.7.4.4 Safety Evaluation

9.7.4.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the D₂O Cleanup System:

None.

9.7.4.4.2 Assessment

The D₂O Cleanup System performs no safety function.

9.7.4.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable criteria:

None.

TABLE 9.7-1

Heavy Water Management

D₂O Inventory

Moderator and Associated Systems	577,600 lb.
Heat Transport and Associated Systems	403,440 lb.
Fuel Handling Systems	9,920 lb.
D ₂ O Management	<u>33,070 lb.</u>
TOTAL	1,024,030 lb.

TABLE 9.7-2

D₂O Supply System Design Data

<u>Item</u>	<u>Data</u>
D ₂ O Supply System	
Storage Capacity	10,000 ft ³
D ₂ O Supply Pumps	
Quantity	2
Type	Canned Centrifugal
Duty	Intermittant
Capacity	100%
Pump Data:	
Fluid	D ₂ O
Flow Rate	125 gpm
Temperature	Ambient
Head	100 ft
Design Pressure	150 psig
Design Temperature	150°F
Motor Data:	
Type	3 Phase, squirrel cage induction motor
Power Supply	Class IV
Construction:	
Materials in Contact with D ₂ O	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 9.7-2 (cont'd)

D₂O Supply System Design Data

<u>Item</u>	<u>Data</u>
D ₂ O Storage Tanks	
Quantity	4
Type	Vertical cylinder with dished ends
Capacity	2500 ft ³
Design Conditions:	
Operating Pressure	0 psig
Operating Temperature	Ambient
Design Pressure	15 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 9.7-3

D₂O Vapor Recovery System Design Data

<u>Item</u>	<u>Data</u>
D ₂ O Vapor Recovery System	
Type	Regenerating Air Dryers
D ₂ O Recovery Capacity	55 lb/hr at 0°F Dew Point
Applicable Code	ASME Section III, Class MC
D ₂ O Vapor Recovery System Condensers	
Quantity	5
Type	Enclosed fin-tube (air to water)
Heat Transferred	90 KW (th)
Tube Side Data:	
Fluid	Chilled H ₂ O
Flow Rate	57.5 gpm
Inlet Temperature	50°F
Outlet Temperature	69°F
Design Pressure	200 psig
Design Temperature	220°F
Shell Side Data:	
Fluid	Air
Flow Rate	1400 ft ³ /hr
Inlet Temperature	300°F
Outlet Temperature	85°F
Design Pressure	20 psig
Design Temperature	300°F

TABLE 9.7-3 (cont'd)

D₂O Vapor Recovery System Design Data

<u>Item</u>	<u>Data</u>
Construction:	
Tube Material	Cu N.
Tube Outside Diameter	0.625 m
Shell Material	Austenitic Stainless Steel
Applicable Codes:	
Tube Side	
Shell Side	ASME Section III, Class MC
D ₂ O Vapor Recovery Collection Pumps	
Quantity	2
Type	Canned Centrifugal
Pump Data:	
Fluid	Downgraded D ₂ O
Flow Rate	25 gpm
Temperature	100°F
Head	50 ft
Design Pressure	150 psig
Design Temperature	150°F
Motor Data:	
Type	3 phase, squirrel cage induction motor
Power Supply	Class IV
Construction:	
Materials in Contact with Fluid	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 9.7-3 (cont'd)

D₂O Vapor Recovery System Design Data

<u>Item</u>	<u>Data</u>
D ₂ O Vapor Recovery Fans	
Quantity	12
Type	Centrifugal
Fan Data:	
Fluid	Air
Flow Rate	4 at 3500 ft ³ /min
	4 at 1100 ft ³ /min
	4 at 175 ft ³ /min
	1 at 440 ft ³ /min
	1 at 550 ft ³ /min
Temperature	Ambient
Design Pressure	20 psig
Design Temperature	100°F
Motor Data:	
Type	3 phase, squirrel cage induction motor
Power Supply	Class IV
Applicable Code:	ASME Section III, Class MC
Vapor Recovery System D ₂ O Collection Tanks	
Quantity	3
Type	Vertical cylinder with dished ends
Design Conditions:	

TABLE 9.7-3 (cont'd)

D₂O Vapor Recovery System Design Data

<u>Item</u>	<u>Data</u>
Operating Press	Atmospheric
Operating Temperature	100°F
Design Pressure	20 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class MC
D ₂ O Vapor Recovery Driers	
Quantity	6
Type	Molecular sieve dessicant
D ₂ O Recovery Capacity	14 lb/hr at 0°F Dew Point
Design Pressure	20 psig
Design Temperature	300°F
Applicable Code	ASME Section III, Class MC

TABLE 9.7-4

D₂O Upgrading System Design Data

<u>Item</u>	<u>Data</u>
Type	Distillation
Feed	Downgraded D ₂ O
Design Feed Flow Rate	630,000 lb/year
Design Feed Isotopic	62% D ₂ O
Design Discard Stream Isotopic	2% D ₂ O
Design Product Isotopic	99.8% D ₂ O
Applicable Code	ASME Section III, Class 3

TABLE 9.7-5

D₂O Cleanup System Design Data

<u>Item</u>	<u>Data</u>
D ₂ O Cleanup System	
Feed Fluid	Recovered Downgraded D ₂ O
Feed Storage Capacity	1700 ft ³
Product Storage Capacity	140 ft ³
Cleanup Process	
Duty	Intermittant
Type	Ion Exchange
Flow Rate	5.6 gpm
Applicable Code	ASME Section III, Class 3
D ₂ O Cleanup Pumps	
Quantity	3 (1 of which is spare)
Type	Canned Centrifugal
Duty	Intermittent
Capacity	100%
Pump Data:	
Fluid	D ₂ O
Flow Rate	5.6 gpm
Temperature	Ambient
Head	30 ft
Design Pressure	150 psig
Design Temperature	150°F
Motor Data:	

TABLE 9.7-5 (cont'd)

D₂O Cleanup System Design Data

<u>Item</u>	<u>Data</u>
Type	3 Phase, squirrel cage induction
Power Supply	Class IV
Construction:	
Materials in Contact with Fluid	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
D ₂ O Cleanup Storage Tanks	
Quantity	6
Type	Vertical cylinder with dished heads
Capacity	283 ft ³
Design Conditions:	
Operating Pressure	0 psig
Operating Temperature	Ambient
Design Pressure	15 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
D ₂ O Cleanup Charcoal Filter	
Quantity	6
Type	Vertical cylinder with dished top and conical bottom
Volume	9 ft ³

TABLE 9.7-5 (cont'd)

D₂O Cleanup System Design Data

<u>Item</u>	<u>Data</u>
Charcoal Capacity	7 ft ³
Design Conditions:	
Flow Rate	5.6 gpm
Direction of Flow	Downward
Operating Pressure	15 psig
Operating Temperature	Ambient
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
D ₂ O Cleanup Ion Exchange Columns	
Quantity	3
Type	Vertical cylinder with dished top and conical bottom
Volume	9 ft ³
Resin Capacity	7 ft ³
Design Conditions:	
Flow Rate	5.6 gpm
Direction of Flow	Downward
Operating Pressure	15 psig
Operating Temperature	Ambient
Design Pressure	150 psig

TABLE 9.7-5 (cont'd)

D₂O Cleanup System Design Data

<u>Item</u>	<u>Data</u>
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

Figure 9.7-1 D₂O SUPPLY SYSTEM

Figure 9.7-2 D2O VAPOUR RECOVERY SYSTEM

9.7-24

Figure 9.7-3 D₂O CLEAN UP SYSTEM

Figure 9.7-4 D₂O UPGRADING SYSTEM

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CHAPTER 10

STEAM AND POWER CONVERSION SYSTEM

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10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Function

The steam and power conversion system provides an overall turbine generator plant arrangement to optimize the energy conversion of the steam from the nuclear steam supply system.

10.2 Description

10.2.1 Main Steam Supply

Steam is supplied from the Reactor Building to the Turbine Building by four main steam lines which are located in a pipe bridge. Main steam lines are seismically qualified to the first anchor outside Containment. Isolation of a main steam line is accomplished by closing the main steam stop valve.

Safety valves and atmospheric steam discharge valves (SDV) are installed on each main steam line. Each SDV is sized for 10% of the main steam flow capacity.

The steam pressure is held constant by varying the reactor power. A turbine bypass system is provided to allow continuous steam to the condenser of up to 70% of rated power steam flow.

10.2.2 Feedwater System

Feedwater is supplied to the steam generators in the Reactor Building from the Turbine Building by four main feedwater lines. Control valves and isolation valves are provided on each feedwater line, with the isolation valves being located just outside containment and a check valve inside containment. Small bypass control valves are provided in parallel with the feedwater control valves. Phosphates are added to the feedwater to control alkalinity for corrosion protection and to convert soluble scale-forming calcium and

magnesium salts into insoluble compounds which are then removed from the steam generator by continuous blowdown. Refer to Figure 10.1-1 for a schematic of the feedwater system.

There are three 50% capacity main steam generator feedwater pumps, complete with electric motor drives, instrumentation and accessories and two 100% main condensate extraction pumps, all of which are on Class IV power, two auxiliary steam generator feedwater pumps supply water to the steam generators by pumping water from the deaerator storage tank. These pumps are powered from a Class III electrical supply. An auxiliary condensate extraction pump, on Class III power replenishes the deaerator storage tank with water from the condenser hotwells, which are replenished with water from the demineralized water storage tank. The entire storage capacity provides decay heat removal for a period of 10 hours. If additional capacity is needed, water from the Emergency Water Supply System could be available.

10.2.3 Turbine Generator

The turbine generator is a single shaft tandem compound 1800 rpm unit. The turbine comprises one double flow high pressure cylinder followed by external moisture separators, live steam reheaters and three double flow low pressure cylinders.

The wet steam from the high pressure turbine exhaust passes through the separators where the water is removed. The steam then passes to reheaters where it is reheated by main steam. The reheating steam is condensed in the reheaters and pumped back to the steam generators.

Extraction points are located on the turbine casings and separators to provide extraction steam to six stages of feedwater heating; two high pressure stages, the deaerator, and three low pressure stages.

All main stop valves, control valves, intercept and emergency stop valves, and the automatic check valves on the extraction lines close automatically in the event of protection system trip.

The turbine governor is of the electrohydraulic type. The control fluid system is independent of the lubricating system and is equipped with two high pressure, ac motor driven, 100% capacity pumps.

The electrical control system is built as an analog computer with solid state circuitry.

The normal governing control system operated to fully close the governing valves at 3% overspeed, and the mechanical hydraulic overspeed trip system is set to operate at 10% overspeed.

The generator is a three phase armature rotating field type. The field is four pole. The stator windings are cooled by low conductivity water circulating through copper conductors around a closed circuit. The stator core is cooled by hydrogen gas flowing through axial and radial ducts in ducts in the core, and the rotor windings are cooled by gas flowing in ducts in the copper conductors of the winding.

Excitation is of the static type, consisting of a solid state automatic voltage regulator controlling a thyristor converter which supplies the generator field via a field circuit breaker and the generator sliprings and brushgear.

The main power output from the generator to the stepup transformer is by means of forced air cooled isolated phase bus duct, with tapoffs to the unit service transformer, excitation transformer and potential transformer cubicle.

10.3 Design Bases

See Table 10.2-1.

10.4 Safety Evaluation

10.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Steam and Power Conversion System:

- GDC 1 - Quality standards and records.
- GDC 2 - Design bases for protection against natural phenomena.
- GDC 4 - Environmental and missile design bases.
- GDC 50 - Containment design bases.
- GDC 51 - Fracture prevention of containment pressure boundary.
- GDC 52 - Capability for containment leakage rate testing.
- GDC 53 - Provisions for containment inspection and testing.
- GDC 54 - Piping systems penetrating containment.
- GDC 57 - Closed system isolation valves.

10.4.2 Assessment

The Quality Assurance Program and recordkeeping required by GDC 1 are only partially complied with.

The seismic capability of the main steam headers will require an evaluation on the basis of the ground accelerations normally assigned to nuclear power plants sited in the USA. Seismic requirements extend the length of the steam headers from the steam generator to the outermost containment isolation valve. In order to comply with GDC 2, the feedwater headers must meet seismic requirements, from the outermost containment isolation valve to the steam generator.

GD4 requires that provisions be made to restrain the high-energy steam and feedwater headers from whipping, if they should break. The restraints should prevent the postulated whipping pipe from damaging components of another system or a structure.

The portion of the main steam headers and feedwater headers and their isolation valves should comply with containment isolation requirements of GDC 50, 51, 52, 53, 54, and 57.

The main steam and feedwater piping systems should be included in the in-service inspection program.

10.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10 CFR 50.
- GDC 2 - No change pending review of design and seismic analysis of the main steam and feedwater headers and their supporting structures.
- GDC 4 - Provide pipe whip restraints at appropriate locations on the high-energy piping systems.
- GDC 50 - The portions of the main steam and feedwater headers that penetrate containment, from the steam generators to the outermost isolation valves, should be designed to ASME III, Class 2 requirements. Material should be selected to meet the requirements for fracture toughness. Isolation valves

should be located outside of containment on both the main steam and feedwater headers. The piping systems should also be included on the inservice inspection program.

GDC 51 - See GDC 50, above.

GDC 52 - See GDC 50, above.

GDC 53 - See GDC 50, above.

GDC 54 - See GDC 50, above.

GDC 57 - See GDC 50, above.

TABLE 10.2-1

STEAM AND POWER CONVERSION SYSTEM DESIGN DATA

Basic Plant Data

Reactor

o Total NSSS Thermal Power (Design) MWt	2,180
o Core Thermal Power (Design), MWt	2,061

Steam Generators (Shell Side)

o Design Pressure, psig	735
o Design Temperature, °F	510
o Expected Operating Conditions	
o Total Steam Flow, lb/hr	8.3×10^6
o Outlet Steam Pressure, psia	631
o Outlet Steam Temperature, °F	500
o Steam Quality, %	99.7
o Feedwater Inlet Pressure, psia	N/A*
o Feedwater Inlet Temperature, °F	368
o Blowdown Flow (continuous), lb/hr	8,000
o Reheater Drains Flow, lb/hr	688,800
o Reheater Drains Enthalpy, BTU/lb	483.68
o Blowdown Enthalpy, BTU/lb	487.9
o Steam Bypass System, % of Full Load	70

Turbine Generator Cycle

o Output, MWe	
o Gross	680
o Estimated Auxiliary Load	60
o Estimated Net Plant	620

TABLE 10.2-1 (cont'd)

Construction

o Main Steam and Boiler Feedwater Piping System

(Inside Containment)

Material

ASTM A-106, GR.B

Code

ASME III, Class 2

(Outside Containment - From First Pipe Anchor)

Material

ASTM A-106, GR.B

Code

ANSI B31.1

o Steam Dump Piping System

Material

ASTM A-106, GR.B

Code

ANSI B31.1

o Auxiliary Feedwater Piping System

Material

ASTM A-106, GR.B

Code

ASME III, Class 3

Note

*N/A - indicates information that is not available in referenced documents.

BOILER STEAM AND FEEDWATER

Figure 10.1-1

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CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT

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11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Function

The heavy water reactor plant design incorporates facilities for permanent storage (on-site burial) and dispersal of all solid and liquid waste. The radwaste equipment, tankage and facilities for handling liquids and solids are designed to handle anticipated operational fluctuations in volume.

11.2 Description

11.2.1 Liquid Waste Management System

This system handles radwastes carried in aqueous streams from the active floor drains, decontamination center, chemical laboratories and laundry. Refer to Figure 11.2-1.

Washrooms and showers are not included since the expected activity is very low and the contamination requires no special handling.

The wastes are collected in a series of sumps and hold-up tanks in the basement of the station. After sampling, the sump contents are either processed directly or mixed with similarly contaminated wastes in the hold-up tanks. Filtration and ion exchange or equivalent processing is provided as required. The hold-up tanks after sampling are also processed with this equipment if necessary. When processing is used, the product is collected in dispersal tanks where it is either reprocessed or dispersed directly to the condenser cooling water system. The liquid waste system is arranged to provide complete flexibility of operation.

For convenience of plant operation and control, active liquid wastes are divided into two categories:

- a) Low activity wastes which are expected to contain less than 10^{-5} microcuries/ml and which are normally discharged to the cooling water discharge channel without treatment to remove activity.
- b) Normal activity wastes which may contain up to 10^{-3} microcuries/ml. These wastes may require processing to remove activity, although normally they can be discharged to the cooling water discharge channel without treatment.

Normally, liquid wastes containing more than 10^{-3} microcuries/ml are not expected. If encountered, they will be processed to reduce activity before discharge. All liquid wastes are discharged at a controlled rate to the cooling water discharge channel. There is an in-line activity monitor in the discharge line.

a) Low Activity Wastes

These wastes originate in such areas as personnel showers, change-rooms, the laundry and the non-active laboratories.

The wastes are collected in any one of three tanks. Each tank is provided with its own pump, mixing eductor and sampling point.

b) Normal Activity Wastes

These wastes originate from the decontamination center, active laboratories, the heavy water upgrading column, excess water from ion exchange resin slurrying, Reactor Building drains, and the rubber goods laundry.

Two liquid waste collection tanks are provided. Each tank is provided with its own pump, mixing eductor and sampling point.

11.2.2 Gaseous Waste Management System

Subsystems are provided to handle gaseous radioactive discharges from the reactor building, spent fuel storage area, decontamination center, D₂O handling area and off-gas system vents.

All active or potentially active gases, vapors, or airborne particulates which occur in the station are monitored and filtered if necessary prior to release to the atmosphere. In particular, active gases which have been vented from the primary system are released to the active ventilation system only after removal of heavy water and storage to permit decay of short half-life isotopes. The tritium produced by activation of the heavy water in the PHTS and moderator D₂O circuits may escape as HTO or a similar form. Unless retained in the D₂O collection system, it will ultimately arrive in the active liquid or gaseous radwaste systems.

11.2.3 Solid Waste Management System

The general sources of solid radwaste are ion exchange resins, contaminated filters, contaminated or activated equipment, miscellaneous contaminated plastics, organic fluids and chemicals.

Demineralized water is used to flush spent resin from the de-deuteration tanks of the primary heat transport and the moderator purification systems, and from the ion exchange columns in the heavy water clean-up system, the spent fuel bay system, the shield cooling system and the radioactive liquid waste system. The slurry is carried by piping to the spent resin storage tank and the excess water collected in the spent resin storage tank is transferred to the radioactive liquid waste system.

Active spent resin is transported in a closed, corrosion-resistant piping system; thereby minimizing exposure to the plant operators while transferring spent resins. Piping material is austenitic stainless steel. The system is designed to ASME Section III Class 2 and Class 3 Code requirements.

The resin transfer system is designed for normal temperature and low pressure operation and connects to separate de-deuteration tanks in the primary heat transport and the moderator purification systems. The resin transfer system cannot be connected to the high pressure or high temperature systems.

The ion exchange resins are stored permanently in underground lined concrete tanks close to the reactor building. Filter cartridges are either buried in flasks on-site or packed into shipping flasks for off-site disposal.

Low activity flammable rags and paper are collected into plastic bags, packed into drums and moved off-site. Organic fluids and chemicals are packed into shielded shipping containers and are moved off-site.

The HWR plant design incorporates an on-site burial area in addition to the ion-exchange resin storage facility. This area has the following facilities:

- a) Concrete trenches for bulky materials with a low specific activity, which are fitted with temporary weather protective covers until a section is completely filled and ready for sealing. Permanent sealing involves back-filling with sand prior to pouring the concrete covers.
- b) Steel-lined concrete holes for material with high specific activity requiring a concrete hatch cover and remote loading.

11.2.4 Effluent Monitoring

11.2.4.1 Liquid Effluent Monitoring

Effluent from the liquid radioactive waste system is monitored continuously, the sample being taken at a point upstream of the confluence with the condenser cooling water flow in order to achieve maximum measurement accuracy.

A final check on the plant effluent is made by continuously pumping a sample from the discharge canal into a drum which in turn is sampled manually. Laboratory instrumentation is used to determine the nature and concentration of the contaminants in the sample. After being sampled, the drum is emptied and the process is repeated. Other instruments monitor the integrity of the moderator heat exchanger and the process water discharge from the Reactor Building.

Class II power is used for the entire monitoring system, i.e., ratemeters, pumps, blowers and recorders.

11.2.4.2 Gaseous Effluent Monitoring

The effluent is discharged through a ventilation exhaust duct which is clear of all building roof heights. Stack effluent monitors are provided to ensure that the limits of permissible releases are not exceeded.

Continuous samples of the effluent are taken to determine concentrations of iodine, wide range gamma and particulates. The signal from each of these monitors is recorded in the control equipment room. A high level signal is annunciated in the control room and will result in a halt to the release of these effluents. Tritium monitoring is carried out by laboratory analysis of bubbler samples.

Class II power is used for the entire monitoring system, i.e., ratemeters, pumps, blowers and recorders.

A separate triplicated gross-beta-gamma monitoring system monitors the duct activity at a point between the above monitors and the Reactor Building exhaust duct dampers. A high activity measurement at any two of the three instruments will close the dampers.

11.3 Design Bases

Not available.

11.4 Safety Evaluation

11.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the design of the Radioactive Waste Management Systems:

GDC 1 - Quality standards and records.

GDC 2 - Design bases for protection against natural phenomena.

GDC 60 - Control of releases of radioactive materials to the environment.

GDC 61 - Fuel storage and handling and radioactivity control.

GDC 63 - Monitoring fuel and waste storage.

GDC 64 - Monitoring radioactivity releases.

11.4.2 Assessment

The safety margin designed into the waste management system cannot be assessed due to the lack of information on design fuel leakage rates and if the system is designed to accept a single active failure.

The practice of utilizing on-site burial of radwaste would be a major licensing issue for a plant sited in the USA.

Much attention is given to restricting releases and providing shielding to as low-as-is-reasonably-achievable; the seismic classification of burial and storage tanks and the quality group classification of the waste management system is not defined.

The design criterion applied in the U.S. is that passive components must be postulated to fail, and further that pressure bearing components be designed to Seismic Category I, and USNRC Quality Group D (Augmented). The system must be designed such that an uncontrolled release due to a passive failure will not result in a total whole body dose (or its equivalent) in excess of 0.5 rem at the site boundary.

11.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

- GDC 1 - Upgrade the Quality Assurance Program to comply with the requirements of this criterion and Appendix B of 10CFR50.
- GDC 2 - No change pending review of the seismic analysis.
- GDC 60 - No change pending detailed review.
- GDC 61 - No change pending detailed review.
- GDC 63 - No change pending detailed review.
- GDC 64 - No change pending detailed review.

11.0-8

LIQUID WASTE MANAGEMENT SYSTEM
FIGURE 11.2-1

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CHAPTER 12

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12.0 RADIATION PROTECTION

12.1 Function

The radiation protection philosophy applied in the design of heavy water reactor plants is intended to limit radiation exposure to plant personnel while performing necessary maintenance on the plant during normal operation and to limit exposure to the general public for both normal and abnormal plant conditions.

12.2 Description

The basic is separated into three zones to provide radiological control.

These zones are:

Zone 1 - a zone which contains no radioactive equipment or components and is normally free of radioactive contamination.

Zone 2 - a zone which does not contain equipment in contact with radioactive materials or fluids and is normally free of contamination. However, the zone possesses the potential to become contaminated through ventilation systems, movement of equipment and/or personnel.

Zone 3 - a zone which contains the main source of radioactive components. This zone has controlled access.

12.2.1 Ventilation

Zone 3 areas are maintained at a slightly negative pressure relative to atmospheric to insure that any airborne contaminants will not migrate to an uncontrolled area. Within Zones 2 and 3, airborne contamination is exhausted by adjustment of ventilation flows to the gaseous waste management system.

Within each zone, shielding is provided commensurate with the level of contamination and is discussed below.

12.2.2 Shielding

The approach to shielding is to reduce individual and group radiation exposure to as low as reasonably achievable.

The shielding design is carried out to provide the following basic conditions:

- a) Radiation fields less than 25 mR/hr at 24 hrs after shutdown in shutdown areas
- b) Radiation fields less than 0.6 mR/hr during reactor operation in normally accessible areas of the reactor building
- c) Radiation fields less than 0.25 mR/hr during reactor operation in areas outside the reactor building

The shielding for radiation protection is considered under four headings:

Primary Shield	-that shielding which attenuates radiation from the reactor.
Secondary Shield	-that shielding which attenuates radiation from the primary coolant.
Auxiliary Shield	-that shielding which attenuates radiation from auxiliary systems, e.g., moderator, fuelling machine, failed fuel detection system.
Supplementary Shield	-additional shielding not categorized as above.

12.2.3 Access Control and Radiation Monitoring

Station procedures are proposed to restrict access within the exclusion area to qualified personnel and to those under escort by qualified personnel.

Wherever possible, use is made of permanent signs and procedures to warn and instruct personnel of any possible danger from radiation. However, there are "Access Controlled Areas" where the radiation hazard is such that entrance must be made only with the knowledge and consent of the control room staff.

A requirement for starting the reactor is that all doors leading to areas of high radiation during reactor operation must be closed and locked. A mechanical interlock system requires that the doors to these areas be closed and locked before the reactor can be brought to power. The keys will be required in the control room for insertion into a key retainer. When all keys are in the retainer, it will be possible to remove the master or reactor control interlock key. Removal of this key will trap all other keys in the retainer, and hence, prevents their use for access to the locked areas.

The master key, when inserted and turned in the solenoid key release interlock, will energize relays to close contacts in the reactor protective systems. When power in the reactor goes above low log power, a contact will de-energize the solenoid in the key release to trap the master key in the interlock.

Above low log power, the master key is unavailable to release the secondary keys for access to controlled areas. Below low log power, removal of the key ensures that the reactor is in the shutdown state.

Entrance to most areas is based on area monitor readings. There are, however, some areas where radiation is directly related to power level, and in these instances an alarm is provided which activates if the power is increased past a specified level while the key is being removed from, or is anywhere other than in the keyboard.

Procedural controls are also proposed consisting of tagging equipment so that machinery cannot be started if it would create a hazard.

Fixed area radiation monitors are permanently installed in areas of potentially dangerous radiation levels to detect the occurrence of radiation hazards and to warn personnel of the presence of high radiation fields. Two setpoints are provided on these monitors which activate as flashing light and audible alarm in the field. The lower set point is adjustable and is for dose control by staff while carrying out jobs in these areas. The higher set point is fixed and will both alarm and be recorded in the control room.

Indication is also provided for loss of signal or loss of power. The field alarm circuitry is tied into the access control system to prevent continuous operation of the alarms when the reactor is at power and shutdown areas are not accessible.

12.3 Design Bases

Incorporated within the description.

12.4 Safety Evaluation

12.4.1 Applicable NRC Criteria

The following NRC General Design Criteria apply to the Radiation Protection of the PHWR:

GDC 61 - Fuel storage and handling and radioactivity control.

GDC 63 - Monitoring fuel and waste storage.

GDC 64 - Monitoring radioactivity releases.

In addition 10 CFR 20 and Appendix I of 10 CFR 50 apply to the levels of radiation limits for operating personnel and for the general public.

12.4.2 Assessment

The requirements for on-line maintenance and fueling necessitate a rigidly controlled health physics program which is normally proposed and developed by the utility or its consultants, and therefore is an owner-oriented responsibility. The amount of internal shielding in the plant is an iterative process during the detailed design phase based on expected levels of radiation as well as operating experience.

The radiation protection features of the heavy water reactor plant meet or exceed equivalent U.S. light water nuclear power plants.

12.4.3 Recommended Design Change

The following design changes are recommended to have the PHWR comply with the listed applicable NRC criteria:

None.

13.0 CONDUCT OF OPERATIONS

This Chapter describes plans for operation of the plant and is beyond the scope of this study.

14.0 INITIAL TEST PROGRAM

This Chapter should provide information on the initial test program for structures, systems, and components of the plant to be performed by plant operators and is beyond the scope of this study.

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15.0 Accident Analysis

15.1 Introduction

A review of the methodology for accident analysis of nuclear power plants in Canada identified the major characteristics of the Canadian approach to reactor safety as discussed below.

All safety systems are analyzed to demonstrate their effectiveness for the purpose required. To this end, a number of single failure accidents are postulated and the suppression or mitigation of their consequences through proper operation of the safety systems is tested.

Another aspect, however, of the effectiveness of the safety systems is investigated by analyzing the extent to which accident consequences are worsened by the absence of a given safety system (other things being the same). This analysis is called coincidental (or dual) failure analysis and is done without regard to the frequency of the events under consideration. These coincident events given (in some cases) larger fission product releases. Since the probability of these failures is one thousand or 80 times lower than that assuming single failure, the Atomic Energy Control Board siting guidelines permit higher radiation dose limits in these cases, as shown in Table 15.1-1.

Coincident failure analysis is somewhat similar to the Maximum Hypothetical Accident concept formally used in the USA to evaluate reactor safety, except that it goes substantially farther. Thus, each postulated process failure is systematically coupled with a failure of each of the three safety systems (Shutdown Systems No. 1 and 2; Emergency Core Cooling and Containment). The resultant matrix of coincident failures is given in Table

15.1-2. The obviously irrelevant cases are indicated by a dash. During the course of the analysis, certain other cases are shown to be trivial.

For the purpose of safety assessment, all systems in the station are categorized as either process or safety systems. Process systems are those required for normal operation whereas safety systems are those provided to limit the release of radioactivity if a failure should occur in a process system. For example:

Process Systems

Primary Heat Transport
Reactor Control
Turbine Plant
Electric Power Supply, etc.

Safety Systems

Shutdown System No. 1
Shutdown System No. 2
Emergency Core Cooling
Containment

The single failure accidents which must be analyzed in the Safety Analysis Report by an applicant in Canada are those listed as process failures in Table 15.1-2.

The current U.S. approach to accident analysis is defined by the USNRC Regulatory Guide 1.70, Revision 2, which specified the standard format and content of safety analysis reports for nuclear power plants in the USA (LWR edition). Chapter 15 of this regulatory guide is appended here as Appendix 15A. The radiation dose limits permitted in the U.S. are those established by 10 CFR 100, as shown in Table 15.1-1.

For the purpose of presenting the Canadian Methodology for accident analysis while following the format established by the Regulatory Guide 1.70, Revision 2, the PHWR single failure accidents given in Table 15.1-1 have been divided into three categories based on assumed accident frequencies, as follows:

1. Faults of Moderate Frequencies	(10 ⁻¹	F	10 ⁻² /year)
2. Infrequent Faults	(10 ⁻²	F	10 ⁻⁴ /year)
3. Limiting Faults	(10 ⁻⁴	F	10 ⁻⁶ /year)

A classification of PHWR accidents is proposed in Table 15.1-3. The accidents listed in this table are those already listed in Table 15.1-2 as well as some other single failure accidents which have been postulated, in addition, based on the PWR accident analysis as performed in the U.S. The accident classification given in the Reference Safety Analysis Report (RESAR-3) for a 1150 MW(e) Westinghouse PWR was used as a model. The division by frequency of occurrence of the postulated PHWR faults was done only by similarity with PWR faults since no information on statistical data of PHWR accident history was available.

Sections 15.2, 15.3 and 15.4 present the evaluation of the events given in Table 15.1-3, as it is performed in Canada. Insufficient information exists, at this time, to carry out a more detailed review.

15.2 Faults of Moderate Frequency

15.2.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition

15.2.1.1 Event Evaluation

No information is available on the analysis of events involving disturbances from wrongful use of reactivity devices at subcritical conditions.

15.2.1.2 Safety Evaluation

Cannot be performed at this time.

TABLE 15.1-1

RADIATION DOSE LIMITS FOR ACCIDENT ANALYSIS

Dose	USA ⁽¹⁾		CANADA ⁽²⁾	
	Whole Body	Thyroid ⁽³⁾	Whole Body	Thyroid ⁽⁴⁾
Individual	25 rem	300 rem	25 rem	250 rem
Population	(5)	(5)	10 man-rem	10 man-rem

NOTES:

- (1) Paragraph 100.11(a) of 10 CFR Part 100, "Reactor Site Criteria", specifies these limits for the whole body dose and thyroid dose from iodine exposure received by an individual located at the Exclusion Area (EA) boundary for two hours after the onset of the postulated fission product release as well as for an individual located at the Low Population Zone (LPZ) boundary during the entire period of the radioactive cloud passage (0 - 30 days).
- (2) Atomic Energy Control Board Siting Guidelines. No exposure times are formally associated with these dose limits.
- (3) Limit for dose received from inhalation of all radioiodines present.
- (4) Limit for dose received from inhalation of ^{131}I only. (The ^{131}I -dose is half that which arises from all radioiodines present and, therefore, for comparable iodine releases the dose limit should be multiplied by 2).
- (5) Population considerations are also specified by 10 CFR 100.11(a) which requires that a proposed site must have a Population Center Distance (PCD) of at least one and one-third times the distance to the outer boundary of LPZ. PCD is defined as the distance from the nuclear reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents. WASH-1235 (October 1974), discusses a methodology for the evaluation of a Site Population Factor for the purpose of considering the population in site comparison.

TABLE 15.1-2

MATRIX OF COINCIDENT FAILURE ACCIDENTS

Process Failures	Safety Systems		
	Shutdown #1 or #2	Emergency Core Cooling	Containment
FUEL AND FUEL HANDLING			
- fuel failures in the core	-	-	
- fuel failures during fuel handling			
ELECTRICAL SYSTEM			
- complete and partial loss of Class 4 power supply		-	-
REACTOR CONTROL			
- reactivity disturbances from wrongful use of reactivity devices at both full and low power		-	-
- loss of primary pressure control			
- loss of secondary pressure control		-	-
REACTOR COMPONENTS			
- flow blockage in a fuel channel			
- failure of primary heat transport system pump circulation		-	-
- loss of shield cooling		-	-
- loss of shutdown cooling	-		-
- loss of service water		-	-
COOLANT SYSTEMS			
- failures in the major pipes of the primary heat transport system	-		
- feeder failure			
- end fitting failure			
- pressure tube failure			
- steam main failure		-	
- loss of feedwater supply		-	

NOTE: The obviously irrelevant cases of coincident failures are indicated by a dash. During the course of the analysis, several other cases are shown to be trivial.

TABLE 15.1-3

PHWR ACCIDENT CLASSIFICATION

1. <u>Faults of Moderate Frequency ($10^{-1} < F^* \leq 10^{-2}$ year^{-1})</u>		
	<u>U.S. Definition</u>	<u>Canada Definition</u>
1.1	Uncontrolled control rod assembly withdrawal from a subcritical condition.	--
1.2	Uncontrolled control rod assembly withdrawal at power.	Reactivity disturbances from wrongful use of reactivity devices at both full and low power.
1.3	Uncontrolled boron dilution.	--
1.4	Loss of external electric load and/or turbine trip.	--
1.5	Loss of normal feedwater.	Loss of feedwater supply.
1.6	Loss of offsite power to station auxiliaries (station blackout).	Complete and partial loss of Class 4 power supply.
1.7	Partial loss of reactor coolant flow.	--
1.8	Excessive heat removal due to the feedwater system malfunctions.	--
1.9	Excessive load increase.	--
1.10	--	Fuel failures in the core.
1.11	--	Loss of primary pressure control.
1.12	--	Loss of secondary pressure control.
1.13	--	Loss of shield cooling.
1.14	--	Loss of shutdown cooling.
1.15	--	Loss of service water.

*Frequency of occurrence per year of plant operation.

TABLE 15.1-3 (cont'd)

2. Infrequent Faults ($10^{-2} < F \leq 10^{-4}$ year $^{-1}$)

<u>U.S. Definition</u>	<u>Canada Definition</u>
2.1 Loss of reactor coolant from small ruptured pipes or from cracks in large pipes, which actuate Emergency Core Cooling.	Feeder/End fitting/Pressure tube failure.
2.2 Minor secondary system pipe break.	--
2.3 Inadvertent loading of a fuel assembly into an improper position.	--
2.4 Complete loss of forced reactor coolant flow.	Failure of primary heat transport system pump circulation.
2.5 Waste gas tank rupture.	--
2.6 Single control rod assembly withdrawal at full power.	--
2.7 --	Flow blockage in a fuel channel.

3. Limiting Faults ($10^{-4} < F \leq 10^{-6}$ year $^{-1}$)

<u>U.S. Definition</u>	<u>Canada Definition</u>
3.1 Major rupture of pipes containing reactor coolant up to and including double ended rupture of the largest pipe in the reactor coolant system (loss of coolant accident).	Failures in major pipes of the primary heat transport system.
3.2 Major secondary system pipe rupture up to and including double ended rupture (rupture of a steam pipe).	Steam main failure.
3.3 Steam generator tube rupture.	--
3.4 Single reactor coolant pump locked rotor.	--
3.5 Fuel handling accident.	Fuel failures during fuel refueling.
3.6 Rupture of a control rod mechanism housing (control rod assembly ejection).	--

15.2.2 Uncontrolled Control Rod Assembly Withdrawal at Power (Reactivity Disturbances from Wrongful Use of Reactivity Devices at Both Full and Low Power)

15.2.2.1 Event Evaluation

The design target for loss of regulation accidents is to maintain the integrity of the primary heat transport system. Specifically, the system is designed to prevent:

- (1) excessive overheating of the fuel which might compromise fuel channel integrity and
- (2) excessive overpressure of the primary heat transport system.

Although centerline fuel melting would not necessarily cause fuel channel failure, it has been conservatively chosen as a safety limit. High fuel rating or high channel power could cause centerline fuel melting. A local power peak of about 7.0 KW/m will melt the fuel at the center, even with normal heat transfer from fuel to coolant. A more extensive power peak could increase channel power and dry out the fuel. Beyond dryout the heat transfer from the fuel decreases and the fuel temperature rises. At some post-dryout power the centerline fuel melts.

A loss of reactivity control which increases reactor power causes both high fuel ratings and high channel powers. Since the onset of dryout occurs before the fuel rating reaches 7.0 KW/m, prevention of dryout was chosen as the target for this accident.

Combining various uncertainties in channel power and a 10% credit for overly-conservative stacking of these uncertainties and anticipated improvements in the critical heat flux correlation, it is calculated that to prevent operation in dryout during a loss of reactivity control accident, the reactor must trip before the power-to-coolant exceeds 119% of full power.

In a loss of regulation accident, shutdown system #1 may be tripped on high neutron flux, on high rate log neutron flux, and on high heat transport system pressure. The locations and the trip setpoints of the ion chambers and of the platinum detectors are chosen to prevent dryout in any channel following loss-of-regulations from several different configurations of the reactivity mechanisms. Shutdown System #2 may be tripped on high neutron flux or on high rate log neutron flux. The locations and the trip setpoints of the ion chambers and of the platinum detectors are chosen to prevent dryout in any channel following loss-of-regulation from several different configurations of the reactivity mechanisms.

The code which is used to model the reactor incorporates point reactor kinetics, a one-dimensional fuel dynamics model with fuel-temperature-induced reactivity feedback, and dynamic models of decay heat, platinum detector response, and transient response of the amplifiers and comparators which operate on the detector signals.

The more pessimistic case of equilibrium fuel is analyzed. The smaller fraction of delayed neutrons and the smaller effect of fuel temperature feedback reactivity leads to a faster power rise for a given input reactivity than is the case for fresh fuel.

Transient results obtained with a computer code for loss of regulation from nominal flux shape at 100% power indicated that no dryout occurs for input reactivity ramps from 0.1 to 2 md/sec regardless of the shutdown system (shutoff rods or poison injection) used to terminate the neutron power transient.

15.2.2.2 Safety Evaluation

All loss of regulation accidents considered are assumed to have fixed flux shapes. This section presents the analysis of loss of regulation accidents from nominal flux shape. No information was available on loss of regulation accidents from different flux shapes which are associated with operating conditions such as startup, shim, or running with a zone controller empty.

15.2.3 Uncontrolled Boron Dilution

15.2.3.1 Event Evaluation

No information is available on the analysis of an uncontrolled boron dilution accident.

15.2.3.2 Safety Evaluation

Cannot be performed at this time.

15.2.4 Loss of External Electric Load and/or Turbine Trip

15.2.4.1 Event Evaluation

No information is available on the analysis of an uncontrolled boron dilution accident.

15.2.4.2 Safety Evaluation

Cannot be performed at this time.

15.2.5 Loss of Normal Feedwater (Loss of Feedwater Supply)

15.2.5.1 Event Evaluation

No information is available on the analysis of a loss of normal feedwater.

15.2.5.2 Safety Evaluation

Cannot be performed at this time.

15.2.6 Loss of Offsite Power to Station Auxiliaries - Station Blackout
(Complete and Partial Loss of Class 4 Power Supply)

15.2.6.1 Event Evaluation

No information is available on the analysis of a loss of offsite power to the station auxiliaries.

15.2.6.2 Safety Evaluation

Cannot be performed at this time.

15.2.7 Partial Loss of Reactor Coolant Flow

15.2.7.1 Event Evaluation

No information is available on the analysis of a partial loss of reactor coolant flow.

15.2.7.2 Safety Evaluation

Cannot be performed at this time.

15.2.8 Excessive Heat Removal Due to the Feedwater System Malfunctions

15.2.8.1 Event Evaluation

No information is available on the analysis of excessive heat removal due to the feedwater system malfunctions.

15.2.8.2 Safety Evaluation

Cannot be performed at this time.

15.2.9 Excessive Load Increase

15.2.9.1 Event Evaluation

No information is available on the analysis of an excessive load increase.

15.2.9.2 Safety Evaluation

Cannot be performed at this time.

15.2.10 Fuel Failures in the Core

15.2.10.1 Event Evaluation

The HWR fuel design is the product of many years of research, development and operation in power reactors. The Zircaloy sheaths have demonstrated their integrity despite the internal fission gas pressure, the power changes and mechanical loads imposed during on-power refueling. Of 50,000 fuel bundles irradiated in HWR's (statistics as of July 1973) only 183 (i.e., 0.36%) have been defected.

The on-power refueling facility enables the removal of defected fuel as soon as it is detected. This minimizes the reactivity in the primary heat transport system components that may need periodic maintenance. Thus, the fission product inventory in the primary coolant is much smaller than that imposed by public safety considerations.

The fuel channel is opened at the end fitting during refueling while the primary heat transport system continues to operate at temperature and pressure. The fueling machines and their control system incorporates many checks and mechanical interlocks to preserve the coolant pressure-boundary should a malfunction occur. Despite the protective measures, there is a low probability that some sequence of failures could cause a breach of the primary heat transport system at the end fitting. At most, all the fuel in one channel could fail and the maximum potential release of I-131 to the containment would be of the order of 10,000 Ci.

15.2.10.2 Safety Evaluation

The inherent characteristics of the PHWR as a pressure tube reactor with on-power refueling of natural uranium fuel elements minimize the consequences of fuel failures in the core. Prompt removal upon detection of defective fuel elements enables operation with a fission product inventory in the primary coolant much smaller than that required by public safety considerations.

15.2.11 Loss of Primary Pressure Control

15.2.11.1 Event Evaluation

No information is available on the analysis of a loss of primary pressure control.

15.2.11.2 Safety Evaluation

Cannot be performed at this time.

15.2.12 Loss of Secondary Pressure Control

15.2.12.1 Event Evaluation

No information is available on the analysis of a loss of secondary pressure control.

15.2.12.2 Safety Evaluation

Cannot be performed at this time.

15.2.13 Loss of Shield Cooling

15.2.13.1 Event Evaluation

No information is available on the analysis of a loss of shield cooling.

15.2.13.2 Safety Evaluation

Cannot be performed at this time.

15.2.14 Loss of Shutdown Cooling

15.2.14.1 Event Evaluation

No information is available on the analysis of a loss of shield cooling.

15.2.14.2 Safety Evaluation

Cannot be performed at this time.

15.2.15 Loss of Service Water

15.2.15.1 Event Evaluation

No information is available on the analysis of a loss of service water.

15.2.15.2 Safety Evaluation

Cannot be performed at this time.

15.3 Infrequent Faults

15.3.1 Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes, which Actuate Emergency Core Cooling (Feeder/End Fitting/Pressure Tube Rupture)

15.3.1.1 Event Evaluation

No information is available on the analysis of ruptures of feeders, end fittings or pressure tubes. The loss of coolant accident following a major pipe rupture is analyzed in Section 15.4.1.

15.3.1.2 Safety Evaluation

Small pipe rupture analysis is important and must be performed in order to evaluate the possibility of rupture propagation to the adjacent pipes (e.g., feeder pipe rupture propagation). Due to the lack of information this safety analysis cannot be evaluated at this time.

15.3.2 Minor Secondary System Pipe Break

15.3.2.1 Event Evaluation

No information is available as the analysis of minor secondary system pipe breaks.

15.3.2.2 Safety Evaluation

Cannot be performed at this time.

15.3.3 Inadvertent Loading of a Fuel Assembly into an Improper Position

15.3.3.1 Event Evaluation

No information is available on the analysis of such an accident.

15.3.3.2 Safety Evaluation

Cannot be performed at this time. However, it appears that the consequences of such an accident would be small in view of the fact that the pressurized heavy water reactor utilizes natural uranium fuel elements.

15.3.4 Complete Loss of Forced Reactor Coolant (Failure of the Primary Heat Transport System Pump Circulation)

15.3.4.1 Event Evaluation

No information is available on the analysis of a complete loss of forced reactor coolant.

15.3.4.2 Safety Evaluation

Cannot be performed at this time.

15.3.5 Waste Gas Tank Rupture

15.3.5.1 Event Evaluation

No information is available on the waste gas tank rupture analysis.

15.3.5.2 Safety Evaluation

Cannot be performed at this time.

15.3.6 Single Control Rod Assembly Withdrawal at Power

15.3.6.1 Event Evaluation

No information is available on the analysis of the single control rod assembly withdrawal at power.

15.3.6.2 Safety Evaluation

Cannot be performed at this time.

15.3.7 Flow Blockage in a Fuel Channel

15.3.7.1 Event Evaluation

No information is available on the analysis of the flow blockage in a fuel channel.

15.3.7.2 Safety Evaluation

Cannot be performed at this time.

15.4 Limiting Faults

15.4.1 Major Rupture of Pipes Containing Reactor Coolant up to and Including Double Ended Rupture of the Largest Pipe in the Reactor Coolant System - Loss of Coolant Accident (Failures in Major Pipes of the Primary Heat Transport System)

15.4.1.1 Event Evaluation

15.4.1.1.1 Introduction

The primary heat transport system, as shown in Figure 15.4-1, consists of two figure-of-eight circuits connected by two 3-inch Sch. 80 lines and 8-inch Sch. 100 line. During normal operation, all of these lines are open but will be closed automatically within 20 seconds of a pipe rupture.

During blowdown following a pipe failure, the coolant pressure drops from about 1450 psig to much lower values (as determined by the Emergency Core Cooling System), resulting in a continuous increase in the outward pressure

differential across the fuel sheath. This is of no consequence if the sheath remains near its operating temperature of 600°F, but an increased sheath temperature leads to a decrease in the Zircaloy strength and ultimate sheath failure.

15.4.1.1.2 Core Voiding Rates

The loss-of-coolant accident empties part of the core coolant which in turn produces positive reactivity and an increase in the reactor power. The core cooling is caused by two mechanisms; flashing of the coolant due to the general depressurization of the primary system and coolant boiling in the core due to continuing heat input from the fuel.

As coolant flow decreases in the core and the steam quality increases, part of the reactor core goes into dryout, thereby increasing the amount of heat stored in the fuel to be removed by the discharging coolant and emergency core cooling.

The power increase is terminated by the shutdown systems. The net effect is a short overpower pulse followed by a power rundown to fission product decay power. Due to the large time constant of UO₂ fuel, the voiding rate is only weakly dependent on the size of this power pulse and is controlled primarily by the break location, break size and reactor condition.

(a) Break Location

The location of a pipe break can be defined relative to the core - either near the inlet end or near the outlet end, where inlet and outlet areas are defined as follows: the pump discharge lines and inlet headers are the large components near the inlet end of the core, and the outlet headers and heat exchangers are the large components near the outlet end.

Pipe failures near the inlet end of the core are more severe than failures near the outlet end with respect to both coolant voiding rate and fuel cooling during the blowdown. Following a large inlet break, the coolant flow initially splits in the downstream core section: the inlet flow reverses while the outlet flow continues in the normal direction. The heat input from the fuel during the flow rundown and reversal caused rapid core voiding. Dryout and the attendant sheath temperature rise also occur early in the blowdown. In the upstream core section, the flow accelerates toward the break and coolant boiling travels back from the outlet end due to depressurization. Voiding is therefore slower than in the downstream core section.

An outlet break causes lower voiding rates than the inlet break for a given break area. The coolant flow in the upstream core section accelerates in the normal direction toward the break and voids in response to falling primary system pressure. Voiding in the downstream core section is caused by reduced inlet flow as coolant is discharged out of the break. There is no flow reversal, and, when dryout does occur, it is much later in the blowdown. The attendant sheath temperature rise is therefore much smaller.

A failure on the suction side of the pumps is a special case. Although this location might be considered near the outlet end of the core, the core first feels the effects of the failure through a deterioration in pump performance associated with boiling at the pump section. The coolant density drops as in the inlet break case, but at a slower rate. The previous discussion describes voiding behavior only in the ruptured circuit. The additional voiding in the unruptured circuit is small since:

- (i) the interconnecting lines are small;
- (ii) the second circuit is not blowing down directly to containment atmosphere, but through a pipe to a partially pressurized circuit; and
- (iii) the two circuits will be completely isolated within approximately 20 seconds of pipe rupture.

(b) Break Size

The voiding rate decreases with break size, gradually in the larger break range and more rapidly in the smaller break range. During most of the blowdown, the coolant discharge rate is limited by choking at the break. The upper limit to the effective break area is twice the flow area of the ruptured pipe. This break area is referred to as the maximum or 100% break.

(c) Reactor Condition

The voiding rate increases with increasing primary heat transport system coolant pressure and reactor power. Higher pressures lead to higher discharge rates and higher powers transfer more heat to coolant, both of which result in faster voiding.

(d) Voiding Rate for the 100% Inlet Header Break

The maximum core voiding rates will result from the largest inlet break during full power operation. The largest component of the primary heat transport system at the inlet end is the inlet header which has a flow area of 1.16 ft^2 . The postulated 100% break area is therefore 2.32 ft^2 .

The transient behavior of the reactor primary heat transport system following a loss-of-coolant accident is simulated with a computer code which calculates the following important transients:

- (1) core voiding
- (2) break discharge rate used for containment pressure transient calculations
- (3) primary coolant pressure transients
- (4) sheath temperature transients

The prediction of transient coolant conditions in the circuit is based on an explicit solution (in one-dimensional space) of the mass energy, and momentum conservation equations. The primary heat transport system is divided into a series of segments and the fuel channel, heat exchanger, pump and associated piping are modeled. Figure 15.4-2 shows the voiding transients in the upstream and downstream sections of the core following a 100% inlet header break.

The decrease in time of the flux-square weighted average density is proportional to the void reactivity increase and is used in a spacial kinetics code along with the reactivity effects associated with fuel temperature changes, shutdown systems and the flux tilt caused by the voiding of only one-half of the core to calculate the neutron power transients in various regions of the core.

15.4.1.1.3 Containment

Containment must withstand the overpressures created by the discharge of hot water from a pipe break and must meet leak tightness requirements at these overpressures.

The maximum containment overpressure will result from the maximum coolant discharge rate which depends on size and location of the break.

The reactor outlet headers and the steam generator inlet pipe carry coolant which is near saturation temperature. The coolant discharge from a break in these pipes will be choked at the sonic velocity of the coolant and the specific discharge rate will be about 50 lbs/sec-in².

The pump suction and discharge pipes and the reactor inlet header carry coolant which is considerably subcooled. The coolant discharge from a break in these pipes will initially be metastable (i.e., coolant discharges through the break before it flashes to saturation) with specific discharge rates of about 80 lbs/sec-in². Obviously then, a maximum break in the pump suction or discharge pipes, or in the inlet header will result in a greater coolant discharge rate than a maximum break in the outlet end pipes. The maximum break in the pumps suction line has been selected as the basis for the containment assessment (equal to twice the flow area of 1.66 ft²).

The integrated coolant discharge predicted by the blowdown code for this break is shown in Figure 15.4-3 and the integrated energy characteristics in Figure 15.4-4.

Although the connecting lines between the two coolant circuits will be closed automatically within 20 seconds of the break, the discharge from the unfailed to the failed circuit during these 20 seconds has been conservatively assumed (for containment pressure calculations) to take place via the interconnects directly to the containment.

The containment pressure rises initially at about 5 psi/sec (see Figure 15.4-5). The dousing spray is signaled to start when the containment pressure reaches 9 psig; there is a 2.75 second delay while the dousing valves

open before the spray begins; the dousing flow builds up over the next 2.25 seconds to the nominal rate of 10,000 lbs/sec with 4 of 6 channels opening. The dousing spray condenses the steam and turns the pressure transient over at 16.9 psig. The design pressure is 18 psig, and the test pressure is 20.7 psig. The containment pressure transient was calculated with digital computer code. The dousing spray will continue until the containment pressure returns to 3 psig. At the end of blowdown, the emergency core cooling will start and the dousing system will cycle on and off until the pressure no longer reaches an "on" set point. The air coolers will then bring the pressure down to 0.5 psig and the filtered air discharge system will lower the pressure to sub-atmospheric.

15.4.1.2 Safety Evaluation

Except for the core voiding rates and containment pressure following a loss of coolant accident there is not enough information on the analysis of this accident to evaluate its radiological consequences.

15.4.2 Major Secondary System Pipe Rupture up to and Including Double Ended Rupture - Rupture of a Steam Pipe (Steam Main Failure)

15.4.2.1 Event Evaluation

The secondary side piping is designed and manufactured to the same standards of material quality and inspection procedures as the primary heat transport piping. Since it operates at lower pressures, the secondary side piping is even less likely to break than the primary side piping. Further, due to the complete isolation between the primary and secondary circuits, there are no sources of radioactivity to contend with in the event of a secondary pipe rupture. This section describes the sequence of events following a guillotine rupture in a 26-inch diameter steam line (total break area of 5.6 ft²).

Figure 15.4-6 is a schematic of the secondary side piping. Four 26-inch diameter pipes direct the steam to a common balance header located outside the reactor building. A computer code is used to calculate the secondary blowdown for a break in one of these four main steam lines. The sequence of events is as follows.

The initial steam discharge rate at the break jumps to about 2.5 times the full power steam flow and then falls off rapidly as the steam drum depressurizes. A fast turbine runback occurs and consequently steam flow to the turbine rapidly falls to zero. The water level in the steam drum oscillates as the feedwater flow control valves respond to steam flow and drum level changes. The level eventually settles down at the zero power set point and the feedwater flow is just sufficient to make up the steam discharge.

The secondary side depressurizes to near atmospheric conditions and stays there; decay heat is removed by steam discharge out of the break. The feedwater train contains several full power minutes of water and thus provides a heat sink until shutdown coolers can be brought in.

On the primary side, the reactor trips on high reactor building pressure. This, combined with the low secondary side temperature, causes rapid depressurization of the primary heat transport system. Very little quality results in the primary heat transport system as the reactor inlet temperature is also falling rapidly during the depressurization due to the high heat transfer rate in steam generators. The pressure control system, having a limited capacity, will be unable to influence the initial primary heat transport system pressure transient but will eventually return the pressure to normal operating values. Therefore, the main steam line break accident does not lead to any fuel cooling problems.

The integrated mass and energy releases via the break into containment during the first 50 seconds of the pipe break are shown in Figure 15.4-7.

The resulting Reactor Building pressure transient is shown in Figure 15.4-8. The pressure reaches a maximum of 17.8 psig at 45 seconds. Further, an end point calculation, assuming that all the primary and secondary stored energy was released into the Reactor Building by the end of blowdown (500 seconds), indicated a pressure of 20.5 psig which is greater than the building design pressure of 18 psig but still less than the test pressure of 20.7 psig.

15.4.2.2 Safety Evaluation

The claim that the secondary side is completely isolated from the primary side and, therefore, radioactivity-free, is not acceptable in the U.S. for the analysis of the Main Steam Line Break Accident. For the purpose of calculating radiological consequences, the LWR analysis of this accident assumes that one steam generator has been affected by a small but continuous leak until the rupture time.

15.4.3 Steam Generator Tube Rupture

15.4.3.1 Event Evaluation

No information is available on the analysis of a steam generator tube rupture.

15.4.3.2 Safety Evaluation

Cannot be performed at this time.

15.4.4 Single Reactor Coolant Pump Locked Rotor

15.4.4.1 Event Evaluation

No information is available on the analysis of a single reactor coolant pump locked rotor.

15.4.4.2 Safety Evaluation

Cannot be performed at this time.

15.4.5 Fuel Handling Accident (Fuel Failures During Fuel Refueling)

15.4.5.1 Event Evaluation

No information is available on the analysis of a fuel handling accident.

15.4.5.2 Safety Evaluation

Cannot be performed at this time.

15.4.6 Rupture of a Control Rod Mechanism Housing - Control Rod Assembly Ejection

15.4.6.1 Event Evaluation

No information is available on the analysis of a control rod ejection assembly ejection.

15.4.6.2 Safety Evaluation

Cannot be performed at this time. It appears, however, that such an accident would have to be postulated non-mechanically since the control rod assemblies are contained inside the calandria, that is, in low-pressure low-temperature environment, and not inside the pressure boundary of the primary heat transport system.

Figure 15.4-1

Figure 15.4-2

Figure 15.4-3
Integrated Mass Release Following LOCA

15.4-14

INTEGRATED ENERGY, BTU $\times 10^8$

Integrated Energy Release Following LOCA

15.4-4

CONTAINMENT PRESSURE TRANSIENT FOLLOWING
MAXIMUM PUMP SUCTION HEADER FAILURE

Figure 15.4-5

15.4-16

Figure 15.4-6
Steam Line Break Model

INTEGRATED MASS AND ENERGY DISCHARGE
FROM A MAXIMUM STEAM LINE BREAK

Figure 15.4-7

CONTAINMENT PRESSURE TRANSIENT FOR
A MAXIMUM STEAM LINE BREAK

Figure 15.4-8

APPENDIX 15-A

**STANDARD FORMAT AND CONTENT
OF
SAFETY ANALYSIS REPORTS
FOR
NUCLEAR POWER PLANTS***

LWR EDITION

SEPTEMBER 1975

U.S. NUCLEAR REGULATORY COMMISSION

*Chapter 15 only.

15. ACCIDENT ANALYSES

The evaluation of the safety of a nuclear power plant should include analyses of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. Such safety analyses provide a significant contribution to the selection of the design specifications for components and systems from the standpoint of public health and safety. These analyses are a focal point of the Commission's construction permit and operating license reviews of plants.

In previous chapters of the SAR, the structures, systems, and components important to safety should have been evaluated for their susceptibility to malfunctions and failures. In this chapter, the effects of anticipated process disturbances and postulated component failures should be examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

The situations analyzed should include anticipated operational occurrences, (e.g., a loss of electrical load resulting from a line fault), off-design transients that induce fuel failures above those expected from normal operational occurrences, and postulated accidents of low probability (e.g., the sudden loss of integrity of a major component). The analyses should include an accident whose consequences are not exceeded by any other accident considered credible so that the site evaluation required by 10 CFR Part 100 may be conducted.

Transient and Accident Classification

The approach outlined below is intended to organize the transients and accidents considered by the Applicant and presented in the SAR in a manner that will:

1. Ensure that a sufficiently broad spectrum of initiating events has been considered,
2. Categorize the initiating events by type and expected frequency of occurrence so that only the limiting cases in each group need to be quantitatively analyzed,
3. Permit the consistent application of specific acceptance criteria for each postulated initiating event.

To accomplish these goals, a number of disturbances of process variables and malfunctions or failures of equipment should be postulated. Each postulated initiating event should be assigned to one of the following categories:

1. Increase in heat removal by the secondary system (turbine plant),
2. Decrease in heat removal by the secondary system (turbine plant),

3. Decrease in reactor coolant system flow rate,
4. Reactivity and power distribution anomalies,
5. Increase in reactor coolant inventory,
6. Decrease in reactor coolant inventory,
7. Radioactive release from a subsystem or component, or
8. Anticipated transients without scram.

Typical initiating events that are representative of those that should be considered by the Applicant in this Chapter of the SAR are presented in Table 15-1. The evaluation of each initiating event should be presented in a separate subsection corresponding to the eight categories defined above. The information to be presented in these subsections is outlined in Section 15.X.X.

One of the items of information that should be discussed for each initiating event relates to its expected frequency of occurrence. Each initiating event within the eight major groups should be assigned to one of the following frequency groups:

1. Incidents of moderate frequency,
2. Infrequent incidents, or
3. Limiting faults.

The initiating events for each combination of category and frequency group should be evaluated to identify the events that would be limiting. The intent is to reduce the number of initiating events that need to be quantitatively analyzed. That is, not every postulated initiating event needs to be completely analyzed by the Applicant. In some cases, a qualitative comparison of similar initiating events may be sufficient to identify the specific initiating event that leads to the most limiting consequences. Only that initiating event should then be analyzed in detail.

It should be noted, however, that different initiating events in the same category/frequency group may be limiting when the multiplicity of consequences are considered. For example, within a given category/frequency group combination, one initiating event might result in the highest reactor coolant pressure boundary (RCPB) pressure while another initiating event might lead to minimum core thermal-hydraulic margins or maximum offsite doses.

Plant Characteristics Considered in the Safety Evaluation

A summary of plant parameters considered in the safety evaluation should be given; e.g., core power, core inlet temperature, reactor system.

Plant Characteristics Considered in the Safety Evaluation

A summary of plant parameters considered in the safety evaluation should be given; e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth and control rod insertion characteristics. A range of values should be specified for plant parameters that vary with fuel exposure or core reload. The range should be sufficiently broad to cover all expected changes predicted for the entire life of the plant. The permitted operating band (permitted fluctuations in a given parameter and associated uncertainties) on reactor system parameters should be specified. The most adverse conditions within the operating band should be used as initial conditions for transient analysis.

Assumed Protection System Actions

Settings of all protection system functions that are used in the safety evaluation should be listed. Typical protection system functions are reactor trips, isolation valve closures, ECCS initiation, etc. The uncertainty (combined effect of calibration error, drift, instrument error, etc.) associated with each function should also be listed together with the expected and maximum delay times.

15.X Evaluation of Individual Initiating Events

The applicant should provide an evaluation of each initiating event using the format of Section 15.X.X (e.g., 15.2.7 for a loss of normal feedwater flow initiating event). As shown in Table 15-1, a particular initiating event may be applicable to more than one category. The SAR sections should be appropriately referenced to indicate this.

The detailed information listed in Section 15.X.X, paragraphs 1 and 2, should be given for each initiating event. However, the extent of the quantitative information in Section 15.X.X, paragraphs 3 through 5, that should be included will differ for the various initiating events. For those situations where a particular initiating event is not limiting, only the qualitative reasoning that led to that conclusion need be presented, along with a reference to the section that presents the evaluation of the more limiting initiating event. Further, for those initiating events that require a quantitative analysis, such an analysis may not be necessary for each of Section 15.X.X, paragraphs 3 through 5. For example, there are a number of plant transient initiating events that result in minimal radiological consequences. The applicant should merely present a qualitative evaluation to show this to be the case. A detailed evaluation of the radiological consequences need not be performed for each such initiating event.

15.X.X Event Evaluation

1. Identification of causes and frequency classification. For each event evaluated, include a description of the occurrences that lead to the initiating event under consideration. The probability of the initiating event should be estimated and the initiating event should be assigned to one of the following groups:

a. Incidents of moderate frequency - these are incidents, any one of which may occur during a calendar year for a particular plant.

b. Infrequent incidents - these are incidents, any one of which may occur during the lifetime of a particular plant.

c. Limiting faults - these are occurrences that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

2. Sequence of events and systems operation. The following should be discussed for each initiating event:

a. The step-by-step sequence of events from event initiation to the final stabilized condition. This listing should identify each significant occurrence on a time scale, e.g., flux monitor trip, insertion of control rods begin, primary coolant pressure reaches safety valve set point, safety valves open, safety valves close, containment isolation signal initiated, and containment isolated. All required operator actions should also be identified.

b. The extent to which normally operating plant instrumentation and controls are assumed to function.

c. The extent to which plant and reactor protection systems are required to function.

d. The credit taken for the functioning of normally operating plant systems.

e. The operation of engineered safety systems that is required.

The effect of single failures in each of the above areas and the effect of operator errors should be discussed and evaluated. The discussion should provide enough detail to permit an independent evaluation of the adequacy of the system as related to the event under study. One method of systematically investigating single failures is the use of a plant operational analysis or a failure mode and effects analysis. The results of these types of analyses can be used to determine which functions, systems, interlocks, and controls are safety related and what readouts are required by the operator under anticipated operational occurrence and accident conditions.

3. Core and system performance.

a. Mathematical model. The mathematical model employed, including any simplifications or approximations introduced to perform the analyses, should be discussed. Any digital computer programs or analog simulations used in the analyses should be identified. If a set of codes is used, the method combining these codes should be described. Important output of each code should be presented and discussed under "results." Principal emphasis should be placed on the input data and the extent or range of variables investigated. This information should include figures showing the analytical model, flow path identification, actual computer listing, and complete listing of input data. The detailed description of mathematical models and digital computer programs or listings are preferably included by reference to documents available to the NRC with only summaries provided in the SAR text.

b. Input parameters and initial conditions. The input parameters and initial conditions used in the analyses should be clearly identified. Table 15-2 provides a representative list of these items. However, the initial values of other variables and additional parameters should be included in the SAR if they are used in the analyses of the particular event being analyzed.

The parameters and initial conditions used in the analyses should be suitably conservative for the event being evaluated. The bases used to select the numerical values that are input parameters to the analysis, including the degree of conservatism, should be discussed in the SAR.

c. Results. The results of the analyses should be presented and described in detail in the SAR. As a minimum, the following information should be presented as a function of time during the course of the transient or accident:

- (1) Neutron power,
- (2) Heat fluxes, average and maximum,
- (3) Reactor coolant system pressure,
- (4) Minimum CHF, DNBR, or CPR, as applicable,
- (5) Core and recirculation loop coolant flow rates (BWRs),
- (6) Coolant conditions - inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam volume fractions,
- (7) Temperatures - maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy,

(8) Reactor coolant inventory - total inventory and coolant level in various locations in the reactor coolant system,

(9) Secondary (power conversion) system parameters - steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, steam generator inventory, and

(10) ECCS flow rates and pressure differentials across the core, as applicable.

The discussion of results should emphasize the margins between the predicted values of various core parameters and the values of these parameters that would represent minimum acceptable conditions.

4. Barrier performance. This section of the SAR should discuss the evaluation of the parameters that may affect the performance of the barriers, other than fuel cladding, that restrict or limit the transport of radioactive material from the fuel to the public.

a. Mathematical model. The mathematical model employed, including any simplifications or approximations introduced to perform the analyses, should be discussed. If the model is identical, or nearly identical, with that used to evaluate core performance, this should be stated in the SAR. In that case, only the differences, if any, between the models need be described.

A detailed description of the model used to evaluate barrier performance should be presented if it is significantly different from the core performance model. The information that should be included is indicated in paragraph 3 of Section 15.X.X, item a.

b. Input parameters and initial conditions. Any input parameters and initial conditions of variables relevant to the evaluation of barrier performance that were not presented and discussed in paragraph 3 of Section 15.X.X., item b, should be discussed in this section. The discussion should present the numerical values of the input to the analyses and should discuss the degree of conservatism of the selected values.

c. Results. The results of the analyses should be presented and described in detail in the SAR. As a minimum, the following information should be presented as a function of time during the course of the transient or accident:

- (1) Reactor coolant system pressure,
- (2) Steam line pressure,
- (3) Containment pressure,
- (4) Relief and/or safety valve flow rate,

(5) Flow rate from the reactor coolant system to the containment system, if applicable.

5. Radiological consequences. This section of the SAR should summarize the assumptions, parameters, and calculational methods used to determine the doses that result from limiting faults and infrequent incidents. Sufficient information should be given in this section to fully substantiate the results and to allow an independent analysis to be performed by the NRC staff. Thus, this section should include all of the pertinent plant parameters that are required to calculate doses for the exclusion boundary and the low population zone as well as those locations within the exclusion boundary where significant site-related activities may occur (e.g., the control room).

The elements of the dose analysis that are applicable to several accident types or that are used many times throughout Chapter 15 can be summarized in this section (or cross-referred) with the bulk of the information appearing in appendices.

If there are no radiological consequences associated with a given initiating event, this section for the event should simply contain a statement indicating that containment of the activity was maintained and by what margin.

Two separate analyses should be provided for each limiting fault. The first analysis should be based on design basis assumptions acceptable to the NRC for purposes of determining adequacy of the plant design to meet 10 CFR Part 100 criteria. These design basis assumptions can, for the most part, be found in regulatory guides that deal with radiological releases. For instance, when calculating the radiological consequences of a loss-of-coolant accident (LOCA), it is suggested that the assumptions given in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," be used. This analysis should be referred to as the "design basis analysis." There may be instances in which the applicant will not agree with the conservative margins inherent in the design basis approach approved by the NRC staff. If this is the case, the applicant may provide an indication of the assumptions he believes to be adequately conservative, but the known NRC assumptions should nevertheless be used in the design basis analysis.

The second analysis should be based on what the applicant believes to be realistic assumptions. This analysis will help quantify the margins that are inherent in the design basis approach. The analysis will also be useful in determining the expected environmental consequences of accidents. This second analysis should be referred to as the "realistic analysis."

The parameters and assumptions used for these analyses, as well as the results, should be presented in tabular form. Table 15-3 provides a representative list of these items. Table 15-4 summarizes additional items that should be provided when dealing with specific types of accidents. When possible, the summary tabulation should provide the necessary quantitative information. If, however, a particular assumption cannot be simply or clearly stated in the table, the table should reference a section or an appendix that adequately discusses the information.

Judgment should be used in eliminating unnecessary parameters from the summary table or in adding parameters of significance that do not appear in Table 15-3 or 15-4. The summary table should have two columns. One column should indicate the assumptions used in the design basis analysis, while the other should indicate assumptions used in the realistic analysis.

A diagram of the dose computation model, labeled "Containment Leakage Dose Model," should be appended to Chapter 15. An explanation of the model should accompany the diagram. The purpose of the appendix is to clearly illustrate the containment modeling, the leakage or transport of radioactivity from one compartment to another or to the environment, and the presence of engineered safety features (ESF) such as filters or sprays that are called on to mitigate the consequences of the LOCA. The diagram should employ easily identifiable symbols, e.g., squares to represent the containment or various portions of it, lines with arrowheads drawn from one compartment to another or to the environment to indicate leakage or transport of radioactivity, and other suitably labeled or defined symbols to indicate the presence of ESF filters or sprays. Individual sketches (or equivalent) may be used for each significant time interval in the containment leakage history (e.g., separate sketches showing the pull-down of a dual containment annulus and the exhaust and recirculation phases once negative pressure in the annulus is achieved, with the appropriate time intervals given).

In presenting the assumptions and methodology used in determining the radiological consequences, care should be taken to ensure that analyses are adequately supported with backup information, either by reporting the information where appropriate, by referencing other sections within the SAR, or by referencing documents readily available to the NRC staff. Such information should include the following:

- a. A description of the mathematical or physical model employed, including any simplifications or approximations introduced to perform the analyses.
- b. An identification and description of any digital computer program or analog simulation used in the analysis. The detailed description of mathematical models and programs are preferably included by reference with only summaries provided in the SAR text.

c. An identification of the time-dependent characteristics, activity, and release rate of the fission products or other transmissible radioactive materials within the containment system that could escape to the environment via leakages in the containment boundaries and leakage through lines that could exhaust to the environment.

d. The considerations of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects taken into account in the evaluation of the results.

e. A discussion of the extent of system interdependency (containment system and other engineered safety features) contributing directly or indirectly to controlling or limiting leakages from the containment system or other sources (e.g., from spent fuel handling areas), such as the contribution of (1) containment water spray systems, (2) containment air cooling systems, (3) air purification and cleanup systems, (4) reactor core spray or safety injection systems, (5) postaccident heat removal systems, and (6) main steam line isolation valve leakage control systems (BWR).

This section should present the results of the dose calculations giving the potential 2-hour integrated whole body and thyroid doses for the exclusion boundary. Similarly, it should provide the doses for the course of the accident at the closest boundary of the low population zone (LPZ) and, when significant, the doses to the control room operators during the course of the accident. Other organ doses should be presented for those cases where a release of solid fission products or transuranic elements are postulated to be released to the containment atmosphere.

TABLE 15-1

REPRESENTATIVE INITIATING EVENTS
TO BE ANALYZED IN SECTIONS 15.X.X OF THE SAR

1. Increase in Heat Removal by the Secondary System
 - 1.1 Feedwater system malfunctions that result in a decrease in feedwater temperature
 - 1.2 Feedwater system malfunctions that result in an increase in feedwater flow
 - 1.3 Steam pressure regulator malfunction or failure that results in increasing steam flow
 - 1.4 Inadvertent opening of a steam generator relief or safety valve
 - 1.5 Spectrum of steam system piping failures inside and outside of containment in a PWR
2. Decrease in Heat Removal by the Secondary System
 - 2.1 Steam pressure regulator malfunction or failure that results in decreasing steam flow
 - 2.2 Loss of external electric load
 - 2.3 Turbine trip (stop valve closure)
 - 2.4 Inadvertent closure of main steam isolation valves
 - 2.5 Loss of condenser vacuum
 - 2.6 Coincident loss of onsite and external (offsite) a.c. power to the station
 - 2.7 Loss of normal feedwater flow
 - 2.8 Feedwater piping break
3. Decrease in Reactor Coolant System Flow Rate
 - 3.1 Single and multiple reactor coolant pump trips
 - 3.2 BWR recirculation loop controller malfunctions that result in decreasing flow rate
 - 3.3 Reactor coolant pump shaft seizure
 - 3.4 Reactor coolant pump shaft break

TABLE 15-1 (Continued)

4. Reactivity and Power Distribution Anomalies

- 4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system), including control rod or temporary control device removal error during refueling
- 4.2 Uncontrolled control rod assembly withdrawal at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (low power to full power)
- 4.3 Control rod maloperation (system malfunction or operator error), including maloperation of part length control rods
- 4.4 Startup of an inactive reactor coolant loop or recirculating loop at an incorrect temperature
- 4.5 A malfunction or failure of the flow controller in a BWR loop that results in an increased reactor coolant flow rate
- 4.6 Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant of a PWR
- 4.7 Inadvertent loading and operation of a fuel assembly in an improper position
- 4.8 Spectrum of rod ejection accidents in a PWR
- 4.9 Spectrum of rod drop accidents in a BWR

5. Increase in Reactor Coolant Inventory

- 5.1 Inadvertent operation of ECCS during power operation
- 5.2 Chemical and volume control system malfunction (or operator error) that increases reactor coolant inventory
- 5.3 A number of BWR transients, including items 2.1 through 2.6 and item 1.2.

6. Decrease in Reactor Coolant Inventory

- 6.1 Inadvertent opening of a pressurizer safety or relief valve in a PWR or a safety or relief valve in a BWR

TABLE 15-1 (Continued)

- 6.2 Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment
- 6.3 Steam generator tube failure
- 6.4 Spectrum of BWR steam system piping failures outside of containment
- 6.5 Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary, including steam line breaks inside of containment in a BWR
- 6.6 A number of BWR transients, including items 2.7, 2.8, and 1.3
- 7. Radioactive Release from a Subsystem or Component
 - 7.1 Radioactive gas waste system leak or failure
 - 7.2 Radioactive liquid waste system leak or failure
 - 7.3 Postulated radioactive releases due to liquid tank failures
 - 7.4 Design basis fuel handling accidents
 - 7.5 Spent fuel cask drop accidents
- 8. Anticipated Transients Without Scram
 - 8.1 Inadvertent control rod withdrawal
 - 8.2 Loss of feedwater
 - 8.3 Loss of a.c. power
 - 8.4 Loss of electrical load
 - 8.5 Loss of condenser vacuum
 - 8.6 Turbine trip
 - 8.7 Closure of main steam line isolation valves

TABLE 15-2

INPUT PARAMETERS AND
INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS

Neutron Power

Moderator Temperature Coefficient of Reactivity

Moderator Void Coefficient of Reactivity

Doppler Coefficient of Reactivity

Effective Neutron Lifetime

Delayed Neutron Fraction

Average Heat Flux

Maximum Heat Flux

Minimum DNBR, CHFR, or CPR

Axial Power Distribution

Radial Power Distribution

Core Coolant Flow Rate

Recirculation Loop Flow Rate (BWR)

Core Coolant Inlet Temperature

Core Average Coolant Temperature (PWR)

Core Average Steam Volume Fraction (BWR)

Core Coolant Average Exit Temperature, Steam Quality, and Steam Void Fraction

Hot Channel Coolant Exit Temperature, Steam Quality, and Steam Void Fraction

Maximum Fuel Centerline Temperature

Reactor Coolant System Inventory (lb)

Coolant Level in Reactor Vessel (BWR)

Coolant Level in Pressurizer (PWR)

Reactor Coolant Pressure

TABLE 15-2 (Continued)

Steam Flow Rate

Steam Pressure

Steam Quality (temperature if superheated)

Feedwater Flow Rate

Feedwater Temperature

CVCS Flow and Boron Concentration (if these vary during the course of the transient or accident being analyzed)

Control Rod Worth, Differential, and Total

TABLE 15-3

REPRESENTATIVE PARAMETERS TO BE TABULATED*
FOR POSTULATED ACCIDENT ANALYSES

- | | Design Basis
Assumptions | Realistic
Assumptions |
|--|-----------------------------|--------------------------|
| 1. Data and assumptions used to estimate radioactive source from postulated accidents | | |
| a. Stretch power level | | |
| b. Burnup | | |
| c. Percent of fuel perforated | | |
| d. Release of activity by nuclide | | |
| e. Iodine fractions (organic, elemental, and particulate) | | |
| f. Reactor coolant activity before the accident (and secondary coolant activity for PWR) | | |
| 2. Data and assumptions used to estimate activity released | | |
| a. Primary containment volume and leak rate | | |
| b. Secondary containment volume and leak rate | | |
| c. Valve movement times | | |
| d. Adsorption and filtration efficiencies | | |
| e. Recirculation system parameters (flow rates versus time, mixing factor, etc.) | | |
| f. Containment spray first order removal lambdas as determined in Section 6.2.3 | | |
| g. Containment volumes | | |
| h. All other pertinent data and assumptions | | |
| 3. Dispersion Data | | |
| a. Location of points of release | | |
| b. Distances to applicable receptors (e.g., control room, exclusion boundary, and LPZ) | | |
| c. χ/Q_s at control room, exclusion boundary, and LPZ (for time intervals of 2 hours, 8 hours, 24 hours, 4 days, 30 days) | | |

*As applicable to the event being described

TABLE 15-3 (Continued)

4. Dose Data

- a. Method of dose calculation
- b. Dose conversion assumptions
- c. Peak [or $f(t)$] concentrations in containment
- d. Doses (whole body and thyroid doses for LPZ and exclusion boundary; beta, gamma, and thyroid doses for the control room)

Design Basis Assumptions	Realistic Assumptions
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TABLE 15-4

ADDITIONAL PARAMETERS AND INFORMATION TO BE PROVIDED
OR REFERENCED IN THE SUMMARY TABULATION FOR SPECIFIC DESIGN
BASIS ACCIDENTS

1. Loss-of-Coolant Accident (Section 15.6.5)

a. Hydrogen Purge Analysis

- (1) Holdup time prior to purge initiation (assuming recombiners are inoperative)
- (2) Iodine reduction factor
- (3) χ/Q values at appropriate time of release
- (4) Purge rates for at least 30 days after initiation of purge
- (5) LOCA plus purge dose at LPZ

b. Equipment Leakage Contribution to LOCA Dose

- (1) Iodine concentration in sump water after LOCA
- (2) Maximum operational leak rate through pump seals, flanges, valves, etc.
- (3) Maximum leakage assuming failure and subsequent isolation of a component seal
- (4) Total leakage quantities for (2) and (3)
- (5) Temperature of sump water vs time
- (6) Time intervals for automatic and operator action
- (7) Leak paths from point of seal or valve leakage to the environment
- (8) Iodine partition factor for sump water vs temperature of water
- (9) Charcoal adsorber efficiency assumed for iodine removal

c. Main Steam Line Isolation Valve Leakage Control System
Contribution to LOCA Dose (BWR)

- (1) Time of system actuation
- (2) Fraction of isolation valve leakage from each release point
- (3) Flow rates vs time for each release path

TABLE 15-4 (Continued)

(4) Location of each release point

(5) Transport time to each release point

2. Waste Gas System Failure (Section 15.7.1)

- a. Activity transfer times to waste gas system components
- b. Number of tanks or other holdup components
- c. Tank volumes
- d. Charcoal bed delay times for Xe and Kr
- e. Seismic classification of tank and associated piping
- f. Decontamination factors of components
- g. Primary coolant volume
- h. Isotopic activity in each system component including daughter products
- i. Time to isolate air ejector
- j. Delay time in delay pipe
- k. Design basis activity measured at air ejector (Ci/sec) including contribution due to activity spiking in coolant

3. Main Steam Line and Steam Generator Tube Failures (Sections 15.1.5, 15.6.3, 15.6.4)

- a. Characterization of primary and secondary (PWR) system (e.g., temperatures, pressures, steam generator water capacity, steaming rates, feedwater rates, and blowdown rates); parameter values should be given for periods prior to, during, and following the accident
- b. Iodine spiking produced by the shutdown and depressurization (ratio of concentration of all iodine isotopes before and after depressurization)
- c. Chronological list of system response times, operator actions, valve closure times, etc.
- d. Steam and water release quantities, and all assumptions made in their computation
- e. Iodine partition factors and their bases

TABLE 15-4 (Continued)

- f. Fuel damage resulting from single rod stuck in combination with the accident
- 4. Fuel Handling Accident (Section 15.7.4)
 - a. Number of fuel rods in core
 - b. Number, burnup, and decay time of fuel rods assumed to be damaged in the accident
 - c. Radial peaking factor for the rods assumed to be damaged
 - d. Earliest time after shutdown that fuel handling begins
 - e. Amounts of iodines and noble gases released into pool
 - f. Pool decontamination factors

The following items should be provided to determine if the calculational methods of Regulatory Guide 1.25 (Safety Guide 25), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," apply:

- g. Maximum fuel rod pressurization
- h. Minimum water depth between top of fuel rods and fuel pool surface
- i. Peak linear power density for the highest power assembly discharged
- j. Maximum centerline operating fuel temperature for the fuel assembly in item i above
- k. Average burnup for the peak assembly in item i above
- 5. Control Rod Ejection and Control Rod Drop Accidents (Sections 15.4.8 and 15.4.9)
 - a. Percent of fuel rods undergoing clad failure
 - b. Radial peaking factors for rods undergoing clad failure
 - c. Percent of fuel reaching or exceeding melting temperature
 - d. Peaking factors for fuel reaching or exceeding melting temperature
 - e. Percent of core fission products assumed released into reactor coolant

TABLE 15-4 (Continued)

- f. Summary of primary and secondary system parameters used to determine activity release terms (see 3 above) from steam line path (PWR)
 - g. Summary of containment system parameters used to determine activity release terms from containment leak paths
 - h. Summary of system parameters and decontamination factors used to determine activity release from condenser leak paths (BWR)
6. Spent Fuel Cask Drop (Section 15.7.5)
- a. Number of fuel elements in largest capacity cask
 - b. Number, burnup, and decay time of fuel elements in cask assumed to be damaged
 - c. Number, burnup, and decay time of fuel elements in pool assumed to be damaged as a consequence of a cask drop (if any)
 - d. Average radial peaking factor for the rods assumed to be damaged
 - e. Earliest time after reactor fueling that cask loading operations begin
 - f. Amounts of iodines and noble gases released into air and into pool
 - g. Pool decontamination factors, if applicable.

16.0 TECHNICAL SPECIFICATIONS

Section 50.36 of 10 CFR 50 requires that an operating license issued by the Commission contain Technical Specifications that set forth the limits, operating conditions, and other requirements imposed on facility operation for the protection of the health and safety of the public.

Specific technical specifications require detailed design and analysis, and as such are beyond the scope of this study.

17.0 QUALITY ASSURANCE

It is the responsibility of the utility or plant owner in the USA to provide for the establishment and execution of a Quality Assurance Program for the plant consistent with the regulations published in 10 CFR 50, Appendix B. This program applies to all activities affecting the structures, systems, and components. These activities include every function from design through operation and maintenance of the facility.

QUALITY ASSURANCE includes those planned and systematic actions necessary to provide adequate confidence that a component or facility will perform satisfactorily in service.

QUALITY CONTROL includes those quality assurance actions which provides a means to control and measure the characteristics of a component, process, or facility to established requirements.

10 CFR 50, provides criteria for the establishment of a Quality Assurance Program that may be implemented in an orderly fashion throughout the plant life. The criteria include the following eighteen areas of concern, which if properly implemented, will provide adequate confidence for the plant's performance:

1. Organization
2. Quality Assurance Program
3. Design Control
4. Procurement Document Control
5. Instructions, Procedures, and Drawings
6. Document Control
7. Control of Purchased Material, Equipment, and Services

8. Identification and Control of Materials, Parts, and Components
9. Control of Special Processes
10. Inspection
11. Test Control
12. Control of Measuring and Test Equipment
13. Handling, Storage, and Shipping
14. Inspection, Test, and Operating Status
15. Nonconforming Materials, Parts, and Components
16. Corrective Action
17. Quality Assurance Records
18. Audits

The experience of the U.S. Nuclear Regulatory Commission and the nuclear industry have demonstrated conclusively the importance of a systematic, disciplined approach to the assurance of quality for the timely and predictable construction and safe, reliable, and economical operation of complex and costly plants. Over the past years, the NRC and the Joint Committee on Atomic Energy have repeatedly emphasized the need for increased attention and more effective application of quality and assurance practices to help prevent costly deficiencies and to better assure the successful achievement of development program objectives.

Level III - Audit, is a documentary review to assure that the QA Program is functioning as planned.

The establishment of the QA Program, the translation of this program into drawings and specifications, and a continuing evaluation of compliance with the intent of the documents is basically an engineering responsibility.

This responsibility includes assigning the proper level of engineering and quality control required for the complexity and importance of the safety-related structure, system, or component.

QA and QC personnel function basically as staff assistance and provide monitoring functions to assure that all requirements of the QA Program and engineering documents are in full compliance with the intent. This function requires a large staff to cope with the inspection, surveillance, and auditing of the contractors and manufacturers involved with a nuclear power plant. Engineering together with the staff assistance of the QA personnel is responsible to assure that quality is first designed into the plant suitable for the functional requirements, the safety requirements, and in accordance with the appropriate codes and standards and, second, to monitor the implementation to assure that the quality of the as-built product is consistent with the original intent.

There is little question that the primary responsibility rests with the Owner or Utility. This responsibility must rest here because in the design and construction, only the Utility can make the final decisions relating to safety margins, design margins, and economics. When the plant is completed, the Utility is the Operator.

A detailed review of the QA Program standards governing the design fabrication and erection of the HWR plants will be made and compared with those imposed by 10 CFR 50, Appendix B. The basis for the review will be the Canadian Standards Association Quality Program Standards:

CSA Standard Z299.1 Quality Assurance Program Requirements

CSA Standard Z299.2 Quality Control Program Requirements

CSA Standard Z299.3 Quality Verification Program Requirements

CSA Standard Z299.4 Inspection Program Requirements

Implementation of 10 CFR 50, Appendix B, for HWR plants erected in the USA will have a significant cost impact due to the fact that it is imposed on:

- Consultants
- Manufacturers
- Suppliers
- Engineering organizations
- Constructors
- Installers
- Owners