Fast Flux Test Facility Final Safety Analysis Report

Amendment 73

D. A. Gantt
Reactor Electrical/I&C Engineering

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Prepared by:

DA Gantt
D. A. Gantt
Principal Engineer

8/31/93
Date

Approved by:

SA Crow
S. H. Crow, Manager
Reactor Electrical/I&C Engineering

8/31/93
Date

Approved by:

JB Waldo
J. B. Waldo, Manager
FFTF Engineering

8/31/93
Date

FFTF FINAL SAFETY ANALYSIS REPORT

RELEASE AND CHANGE CONTROL RECORD

ACTION	DATE	CHANGE	APPROVAL
Amendment 72 (Continued)	08/15/92	<u>Replace/add pages:</u>	<u>Engineering Change Notices:</u>
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		17-587	ECN 158535
		17-592	ECN 158274
		17-620 thru -621	ECN 170389
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		19.3/4-17	ECN 158280
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		20.4-1	ECN 167101
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		20.4-6	ECN 132542, 167102
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		20.4-11	ECN 132542
		20.4-19	ECN 132542
		20.5-1	ECN 132542
		20.5-2	Text Rotation
		20.5-2.1	ECN 132542, New Page
		20.5-5 thru -6	ECN 132542
		20.5-7 thru -8	Text Rotation
		20.5-9	ECN 132542, 167102, Text Rotation
		20.5-10	ECN 132542, Text Rotation
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		20.5-13 thru -14	Text Rotation
		20.5-14.1	ECN 132542, Text Rotation
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		20.5-15	ECN 132542
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		<u>Appendix B</u>	
		B.5-44	ECN 170339
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		RCCR-86 thru -87	New page
		F-3.2-1	187180
		5B.2-13	185156
		5B.2-14	Editorial
		5B.4-40	136350
		7.0-v thru -vi	Update
		7.0-x	Update
		7.3-33	172854
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		7.3-36 thru -37	172854
		7.3-38	Text rotation
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		9.8-2	170463
		9.8-6	Text Rotation
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		9.8-9	175052
		9.8-10	Text rotation
		9.8-14	136350
		9.8-21	136350
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		12.1-37	Editorial
		12.1-37.1	172881
		12.1-38 thru -38.1	Text rotation
		12.1-39 thru -40	172881
		T-12.1-20.2	172881
		T-12.2-6.1	172881
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		13.3-49	136350
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		17-xvii thru -xix	168475
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		17-23 thru -25	141842
		17-39	165580
		17-41	Text rotation
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		17-105	141842
		17-109	141842

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		17-232.1	165580
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		17-290	141842
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		17-334 thru -334.5	165580
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		17-339	165579
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		17-664	165580, New page
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		<u>Appendix C:</u>	
		C.11-6 thru -8	172854
		C.11-8.1	172854, New page
		<u>Appendix F:</u>	
		F-i	Update
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		F-iv.i thru -iv.ii	Update
		<u>Replace/add pages:</u>	
		F-v	Update
		F-vi	Text rotation
		F-vii	Update
		F-viii	Text rotation
		F-ix thru -x	New page, Text rotation
		F.3-7 thru -12	165580
		F.3-13 thru -17	165580, New page
		T-F.3-3 thru -5	165580
		F-F.3-9 thru -13	165580, New page
		F.A-iii	Update
		F.A-15	165579
		F.A-16	165579, New page

SECTION 3.2 FIGURES

The Seismic Category I Piping and Instrument Drawing (P&ID) Index is given on the following pages. The drawings identify, by means of red lines, the Seismic Category I structures, components, piping, and instruments, as well as active components that are required to perform a safety function. Revision numbers are current as of July 1988. Although the drawings in the index are designated with the suffix RL, the released drawings are designated with the suffix S.

All 89 listed drawings are contained in a separate document that has been published for FFTF Plant use. A limited number of these documents are available for those FSAR holders who have a need to know the details provided by these "red line" P&IDs. A copy may be obtained by sending a written request with your FSAR copy number identification to the address below.

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Drawing Number	Drawing Revision	Title	System
H-4-XXXXRL	1	P&ID Redline Drafting Guidelines	-
H-4-11007RL	24	P&ID, Radioactive Liquid Waste	24
H-4-11019RL	27	P&ID, Main Heat Transport, Loop No. 1	51
H-4-11020RL	14	P&ID, Main Heat Transport, Secondary Loop, DHX Loop No. 1	51
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H-4-11035RL	28	P&ID, IDS and Primary Sodium Storage Processing System	81
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H-4-11059RL	21	P&ID, Primary Argon Cover Gas Sampling, CLS No. 1 & 2	85
H-4-11060RL Sh. 1	27	P&ID, Primary Argon Cover Gas Sampling, HTS	85
H-4-11075RL	22	P&ID, Instrument Air Service Piping System	23
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H-4-11080RL	25	P&ID, Cooling in Containment, Service Piping	23
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FIGURE 3.2-1. Seismic Category I Piping and Instrument Drawing (P&ID) Index, Rev. 5, August 1988.

5B.2.3.7 Dump Heat Exchanger

The dump heat exchangers (DHX) are sodium-to-air heat exchangers that transfer heat from the secondary loop sodium to the ambient air. There is one DHX unit (4 modules) in each of the three secondary loops. Each DHX module is equipped with a fan with variable inlet vanes, fan drive motor, fine and coarse control dampers, isolation gates, controls (manual and automatic), tube bundle, oil-fired preheater, ducting, stack and airflow turning vanes. The nominal thermal rating of a module is 33.3 MWt with 90 °F air at initial operating conditions. The modular design provides flexibility of operation (particularly during startup, shutdown, and decay heat removal) and minimizes damage and loss of plant availability in the event of a failure of a single DHX module (Figures 5B.2-14 and 5B.2-15).

The design features that have been incorporated into the DHX to meet the system operating requirements are as follows:

- A. Maximum sodium pressure drop of 60 psi at design flow rate
- B. Fully drainable and self-venting
- C. The geometric center of the DHX tube bundle is a nominal 30 ft above the IHX geometric center (as built dimension is 29 ft 7 3/4 in. — the adequacy of the thermal driving head was confirmed by the Acceptance Test Program before reactor startup)
- D. Sufficient air natural draft through the tube bundle to dissipate a minimum of 10% of design power at initial operating conditions
- E. Airflow shutoff devices and insulation capability to limit the heat loss to approximately 0.167 MWt per module (at 400 °F sodium inlet and -10 °F outside ambient)
- F. Actuators for the variable inlet vanes, outlet dampers, and isolation gates are provided with hand-operated overrides and are located so that they are protected from damage by a sodium fire in a module or adjacent modules
- G. A failure, including a fire in a module, will not affect the operational effectiveness of the adjacent modules.

The DHX tube bundle is made up of a bank of 66 four-pass serpentine tubes, located between horizontal upper and lower headers. The four-pass serpentine tubes consist of 30-ft-long finned sections joined by bare U-tube return bends. Suitable baffling at the sides and ends minimizes bypass of air. Turning vanes are provided to ensure uniform airflow over the tube bundle. The design and maximum operating conditions for the DHX tube bundle are listed in Table 5B.2-6.

The variable inlet vanes, isolation gates, and control dampers are used to limit heat loss during heatup and low-power conditions. The isolation gates and control dampers are used to isolate the air side of a DHX module in the event of a sodium fire. Electrical interlocks prevent closure of the DHX isolation gates unless the DHX fan drive motor is deenergized. The isolation gate actuators are limited in size so that the gates cannot close against a pressure greater than 1 1/2 in. of water. Since the normal DHX fan has a head of at least 5 to 6 in. of water, inadvertent closure of the DHX isolation gates is prevented.

The air supply system consists of a double width/double inlet centrifugal fan with airflow control provided by a variable speed coupling to vary fan speed and by variable inlet guide vanes. At the DHX thermal design conditions, an airflow of 2.25×10^6 lb/hr with an air side pressure drop of 11.0 in. of water requires 1,170 fan horsepower (hp). The fan is driven by a 1,250 hp electric motor. Each module has a stack above the tube bundle that is sized to ensure sufficient natural air draft to remove at least 10% heat load (3.3 MWt) at initial operating conditions. The DHX is designed to quickly shut off airflow by means of vane closure while the fan is coasting down following a scram signal. This airflow reduction ensures that sodium outlet temperature changes to not exceed in severity the thermal transients for which DHX outlet nozzle and secondary cold leg equipment and piping are designed.

A separate indirect oil-fired heater is provided for each module to preheat the DHX tubing before sodium fill and also to establish or maintain a temperature of 400 °F during standby when sufficient decay or pump heat is not available.

A 0.5 to 0.75-inch long through-wall crack in the HTS piping may leak 50 to 150 grams/hr. This is below the reliable threshold for the continuity detectors (see Table 5B.4-2) but sufficient for detection by the inerted cell aerosol monitors. Appendix B discusses sizes of flaws, crack leak rates, critical crack lengths, and detector capabilities.

5B.4.2.3.6 Sensitivity and Operability Tests

From extensive development programs conducted at Atomics International, the Liquid Metal Engineering Center⁽³⁴⁾ and Westinghouse Advanced Reactors Division⁽³⁵⁾ tests have provided the design values for sensitivity of leak detection systems (see Table 5B.4-2). Later tests are discussed in Appendix B. The operability of the leak detection systems will be verified as part of the plant acceptance tests for plant instrumentation. Refer to Chapter 14 for a description of the plant acceptance tests.

5B.4.2.3.7 Confinement of Leaked Coolant

The Reactor Vessel, primary pumps, and IHXs are surrounded by guard vessels which serve to contain sodium and prevent sodium from being sprayed on nearby components or instruments in the very unlikely event of a leak. Primary piping outside guard vessels and secondary piping is encased in thermal insulation. A sheet metal thermal reflector next to the pipe wall would catch leaking sodium, thus protecting nearby equipment. The air ducting panels surrounding the DHX tube bundle would prevent hot sodium from striking adjacent equipment. The guard vessels, piping insulation, and DHXs are described in 2.6.7, 2.9, and 2.7.3 of SDD-51. Any sodium that reaches the floor is further confined as follows:

1. The primary cells are equipped with hot liners which can contain sodium spills without any building structural damage
2. The secondary system cells and pipeways located in the HTS Service Buildings and the DHX enclosures (except for the cruciform area between the DHX modules) are equipped with segmented catch basins (spill pans).

5B.4.2.4 Fire Protection

Design provisions for controlling and minimizing the effects of sodium fires are described below:

- A. Primary sodium-containing components and piping are installed inside steel-lined cells. The cells contain inert gas (nitrogen) during normal reactor operation.
- B. Sodium-containing piping and components are provided with sodium leak detectors to warn of the incidence of sodium leakage and a potential fire hazard.
- C. To the maximum extent feasible, all secondary sodium piping and components are installed in cells or pipeways that can be sealed and rendered gas tight (except in the DHX cruciform areas) so that fires would be self-extinguishing. Nitrogen lines are available to facilitate inerting each area. These cells and pipeways contain segmented sodium catch pans to contain any leaking sodium and prevent its spread.
- D. Each DHX module is equipped with isolation gates and control dampers that can be closed in the event of a sodium leak to suppress and minimize the effects of a fire. After isolation, modules may be flooded with nitrogen for fire control. Sodium catch basins are provided at the lower end of each module to receive and contain all sodium that can reasonably be expected from any plausible leak. The DHX atmosphere is periodically monitored (when shut down) for sodium aerosol.
- E. Some detectors are provided in the secondary sodium areas, except in DHX modules, to signal the presence of a sodium fire. Smoke detectors are provided on the H&V (System 25) inert gas cooler blower unit to signal the presence of fire when the primary sodium areas are deinerted.
- F. Powder or granular fire extinguishment is available for hand application to control small fires that may occur during normal operation. When the inerted cells are deinerted, similar equipment is available for placement in the cells.

The fire protection plan is based on the early detection of sodium leaks.

Without sodium leaks, sodium fires are not possible.

Spark plug type leak detectors are provided on all HTS sodium equipment and piping in containment. An aerosol leak detection system is provided for the Reactor Guard Vessel annulus and for the reactor cavity and HTS cells.

Ionization type smoke detectors are provided in

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7.3.3.4.3 Level 3

The RVIS sensors could be destroyed in an HCDA. The system is not required to be operational following a hypothetical event. The sensors are located in dry thimbles and do not form a leak path through the reactor coolant boundary.

7.3.4 FUEL FAILURE MONITORING SYSTEM (SDD-94)

7.3.4.1 Design Bases

7.3.4.1.1 Functions and Requirements

A. Functions

The Fuel Failure Monitoring (FFM) System performs reactor (SDD-31) fuel and absorber pin failure detection, characterization, location, and verification functions and closed loop (SDD-61) fuel failure detection, characterization, and verification functions for all reactor modes.

Exceptions:

- Refueling - Primary Argon Cover Gas Sampling System, which delivers the reactor cover gas to the FFM, is normally shutdown during refueling to prevent blocking the system's vapor trap assembly with sodium oxides. Reactor cover gas chemistry is monitored during refueling by the refueling chromatograph and radiation detector.
- Shutdown - The Reactor Fission Gas Monitoring System cannot detect fuel failures after the reactor has been shut down for greater than 65 days because of decay of the radioactive isotopes in the fuel to below detectable limits.
- Modes 3 & 4 - 8 hours after shutdown, the quantity of delayed neutron precursors in the fuel is less than that required to produce a detectable signal on the Delayed Neutron Monitoring System.

Specific functions of the Fuel Failure Monitoring System include the following:

1. Detecting pin failures in driver fuel assemblies, absorber assemblies, open loop test assemblies, and closed loop test assemblies

2. Locating driver fuel assemblies, absorber assemblies, and open loop test assemblies containing failures
3. Characterizing failures to identify the degree of fuel and sodium coolant interaction, to estimate the burnup and linear power of failed fuel, and to determine the amount of gas escaped or escaping
4. Collecting and transmitting data on fuel and absorber failures to a central control room (SDD-90) and the DDH & DS System (SDD-91) as required for displays, alarms, operational control, and documentation
5. Verifying that a suspect assembly does or does not contain a failure.

B. Design Requirements

1. Process Requirements--Major process requirements for the FFM System are dependent upon process requirements of interfacing systems. Major requirements imposed upon the FFM System are as follows:
 - a. The FFM System shall be designed to be compatible with the interface process requirements listed in SDD-94, Table 1-I
 - b. The FFM System shall be designed to meet or exceed the process requirements listed in Table 7.3-6.
 - c. When applicable, the FFM System components and materials shall be compatible with the following fluids:

(1) Primary systems coolant	-sodium
(2) Primary systems cover gas	-argon
(3) Equipment cell atmospheres	-nitrogen/air
(4) Readout and Control Instrumentation	-air
2. General
 - a. The FFM detection and location systems initially provided in the FFTF shall be designed for reactor operation with a background of nonvented and nonfailed fuel.

- b. Provisions shall be incorporated in the FFTF systems designs to permit modifications to the detection and location systems for eventual operation with as much as 1% of the core fuel failed and leaking fission gases.
- c. The driver fuel, absorber assembly, and open loop test fuel shall be monitored by common detection and similar location systems.
- d. The goal of the FFM System shall be to located fuel and absorber cladding failures without removing assemblies from the reactor vessel.
- e. The FFM detection and location systems shall be designed such that no single component failure results in contact of air with the reactor primary sodium or causes leakage of reactor or closed loop cover gas to work areas.

7.3.4.1.2 Design Parameters

A. Design Life

All sensors and their leadout wires shall be capable of withstanding the sensor current, environment, temperature, pressure, vibration, and radiation flux conditions expected at their locations to provide design goal lifetimes of at least one year. A target service life of 20 years shall be used as a basis for the design for the rest of the system.

7.3.4.1.3 Service and Transient Conditions

A. Environmental Conditions

In addition to the requirements of Table 7.3-6, sensors and local wiring and signal conditioning equipment shall be designed and installed such that they can withstand the extremes of conditions imposed by the process being measured and the environmental condition in the area of their installation.

7.3.4.1.4 Documentary Criteria

A. NE Standards

The NE Standards applicable to the system are listed in SDD-94. A specific FFM Standard is NE C 15-11T, BF₃ Gamma Tolerant Neutron Detector Tubes.

7.3.4.2 Design Description

The FFM System provides the equipment necessary to detect, characterize, locate, and verify fuel and absorber pin failures. This equipment is illustrated in Figure 7.3-7 and includes sensors, associated signal conditioning equipment, read-out equipment, and annunciator units necessary to accomplish the following:

1. Monitor the reactor cover gas for gaseous fission product activity and indicate abnormal activity levels
2. Monitor the reactor cover gas for activated tag gas released from failed absorber assemblies and indicate high activity levels
3. Monitor the reactor primary sodium coolant for delayed neutrons and indicate abnormal activity levels
4. Monitor each Closed Loop System (CLS) for fission gas and delayed neutrons and indicated high activity levels
5. Locate failed fuel and absorber assemblies
6. Provide for sampling and analysis of reactor and closed loop primary sodium and cover gas
7. Provide data on failure occurrence, failure characteristics, and verify failure location to plant operations personnel.

The FFM System is not a total, integrated system. The system comprises several independent subsystems and analytical procedures providing data on failure occurrence, character, and location to plant operations personnel. Operator response to fuel failure alarms is specified in the operating procedure outlines in Section 4.0 of SDD-94. Since the reactor plant is designed to operate safely with as much as 1% of the core fuel leaking fission gases, fuel failure signals from FFM equipment are not used or needed by the PPS (SDD-99) to initiate reactor scrams.

7.3.4.2.1 Reactor Cover Gas Monitoring System

The Reactor Cover Gas Monitoring System monitors the reactor cover gas for gaseous fission products, which are normally present as "tramp" background, and for gaseous fission and activation products, which escape from failed fuel and absorber pins, respectively. The purposes of this system are (1) to detect fuel and absorber pin failures and (2) to help characterize the

failures as to type and degree. The system is composed of three major subsystems: the Reactor Cover Gas Sampling Subsystem and two (identical in function) Reactor Fission Gas Monitoring Subsystems.

The Cover Gas Sampling Subsystem provides a controlled flow of reactor cover gas through a small, charcoal-filled sample column. The sample column concentrates the radioactive xenon and krypton gases relative to ^{23}Ne and ^{41}Ar .

The Fission Gas Monitoring Subsystems scan the sample column with a pure Germanium (Ge) diode gamma detector. Three specific fission gas isotopes (^{133}Xe , ^{135}Xe , and $^{85\text{m}}\text{Kr}$) and one activation gas isotope (^{125}Xe) are individually monitored. Absorber pin failures are detected by measuring ^{125}Xe , formed by the activation of ^{124}Xe , which is a constituent of the tag gas used for absorber pin failure location. Alarms in the Reactor Control Room are activated in the event of an abnormal activity level for any of these isotopes.

The Fission Gas Monitoring Subsystems analyze data collected from the gamma detector and the FGMs module control panels during a fuel or absorber pin failure to determine the basic characteristics of the failure (e.g., magnitude and burnup).

The Reactor Cover Gas Monitoring System equipment is located in Cells 495E & B and Operating Area 485 of HTS-South, with readout panels and alarms in the Reactor Control Room.

7.3.4.2.2 Absorber Failure Monitoring

Cover gas monitoring for ^{125}Xe (an activation product of the tag gas isotope, ^{124}Xe) is accomplished by the Reactor Cover Gas Monitoring System. An increase in counts from ^{125}Xe in the absence of increases in counts from fission gas radioisotopes, ^{133}Xe and ^{135}Xe , is an indication of an absorber pin failure.

7.3.4.2.3 Reactor Delayed Neutron Monitoring System

The Reactor Delayed Neutron Monitoring System monitors the primary coolant for delayed neutrons from precursors released from failed fuel. A detector configuration is located on each of the three primary HTS hot leg pipes. Each

configuration uses a neutron detector tube arranged to scan a solid angle of the 28-in. OD pipe. The signals from the tube are conditioned electronically to provide a recordable signal that activates an annunciator in the control room whenever the precursor activity in any of the HTS pipes exceeds a predetermined level. A delayed neutron annunciation accompanying or following a rise in the fission gas activity indicates that cladding failure has exposed fuel to the sodium coolant. Signals from the detectors are fed to DDH&DS.

The detectors are located adjacent to the HTS pipes within the HTS cells. This system uses BF_3 neutron detector tubes.

7.3.4.2.4 Closed Loop Cover Gas Monitoring System

Each CLS (SDD-61) will be provided with an independent Cover Gas Monitoring System. These systems will monitor the cover gas in their respective loops for escaped gaseous fission products from failed test assemblies.

The closed loops will be provided with continuous online Ge diode detector gamma ray monitoring systems that are similar in design to the Reactor Cover Gas Monitoring System. The systems will monitor the primary CLS cover gas for the purpose of (1) detecting test assembly failures and (2) providing the entire gamma spectrum, resulting from the failure, to the Fission Gas Monitoring Subsystems.

The Closed Loop Cover Gas Monitoring Systems for CLS No. 1 and No. 2 will be installed at reactor startup. The other two monitoring systems will be installed on a schedule consistent with the installation of CLS No. 3 and No. 4.

The Closed Loop Cover Gas Monitoring Systems will be located in Cells 495 A, B, C, and D and Operating Area 485 of HTS-South. Readout panels and alarm circuits are located in the Reactor Control Room.

7.3.4.2.5 Closed Loop Delayed Neutron Monitoring Systems

Individual delayed neutron monitors consisting of BF_3 detectors in an appropriate moderator and shielding assembly will be installed on each of the main closed loop hot leg pipes (SDD-61). The Closed Loop Delayed Neutron Monitoring Systems for CLS No. 1 and No. 2 will be installed at reactor startup. The other two closed loop monitoring systems will be installed on a schedule consistent with the installation of CLS No. 3 and No. 4. Delayed

neutron monitors provide for continuous monitoring of the sodium flowing from the test assembly. If the precursor activity exceeds a predetermined level, the closed loop tests could be shut down by the operator to protect the closed loops from excessive fission product contamination. This also requires reactor shutdown.

7.3.4.2.6 Failure Characterization System

The purpose of the Failure Characterization System is to provide reactor operating personnel with information as to the nature and severity of a fuel or absorber pin failure. The Failure Characterization System categorizes fuel and absorber pin failures according to type, magnitude, possible location, linear power, and burnup. This system uses data inputs from the following:

1. Reactor Fission Gas Monitoring Subsystem
2. Gas Tag Sampling Trap Samples
3. Post Failure Sampling and Analysis
4. Reactor and Closed Loop Delayed Neutron Monitoring
5. Post Failure Handling, Examination, and Analysis.

Online, continuous systems enable plant personnel to monitor the condition of a failure under reactor operating conditions. Since continued operation with some failed fuel is an eventual goal of the FFTF, it is important that information be available from continuous monitoring systems. These data will be used to determine the proper corrective action, if any, that needs to be taken after a failure is detected.

7.3.4.2.7 Failure Location System

The purpose of the Failure Location System is to identify the incore location of failed fuel or absorber pin assemblies with a minimum of reactor downtime. The following systems and procedures will be available to locate the failed assembly:

1. Gas Tagging Failure Location System
2. Failure Characterization System
3. Reactor Power Cycling Procedures.

The Gas Tagging Failure Location System is the prime system for locating failed fuel and absorber assemblies. The system is based on loading a unique gas tag consisting of a mixture of enriched stable xenon and krypton isotopes into all pins of each fuel and absorber assembly. If a pin fails, a portion of the tag gas will escape to the reactor cover gas to be sampled by the Gas Tag Sampling Trap. The identity of the failed assembly is determined by mass spectrometer analyses of the concentrated gas tag sample.

The Failure Characterization System supplies information, such as linear pin power and burnup estimates, based upon data obtained from the online spectrometer-computer, from cover gas grab samples, and mass spectrometer analyses of

plant operational availability but cannot impair the ability to shut down the plant and maintain safe shutdown. Further discussion of loss of ventilation is contained in Section 3.10.

7.3.9.4.2 Level 2

No PPS functions are required of, or provided by, this system.

7.3.9.4.3 Level 3

This system is not required to mitigate the consequences of the HCDA.

7.3.10 PLANT FIRE PROTECTION INSTRUMENTATION SYSTEM (SDD-93/7)

7.3.10.1 Design Basis

7.3.10.1.1 Functions and Requirements

A. Function:

The Plant Fire Protection Instrumentation System provides instrumentation for the Plant Fire Protection System (SDD-26) to perform the following functions:

1. Monitor and control the fire water supply system
2. Detect and alarm fire conditions
3. Monitor and control automatic extinguishing equipment
4. Monitor and control charging of pre-action extinguishing systems
5. Monitor and control the Aqueous Film Forming Fire Suppression System.

B. Design Requirements:

The Plant Fire Protection Instrumentation System is required by System 26 to perform the control and monitoring functions described above during normal plant operations. No portion of the Plant Fire Protection Instrumentation System is required for operation of the PPS (SDD-99).

7.3.10.1.2 Design Parameters

General coverage of this topic may be found in paragraph 7.3.6.1.2, Design Parameters.

7.3.10.1.3 Service and Transient Conditions

Complete specification of service and transient conditions is given in SDD-93/7 and is summarized below.

A. Electrical Requirements:

Fire detection and alarm equipment and extinguishing control equipment shall be powered from self-contained batteries continuously maintained at design charge by the normal power system.

The fire water motor driven pump control system shall be powered from the emergency power system.

B. Atmospheric Environmental Requirements:

Instruments are designed and installed such that they can withstand the extremes of conditions imposed by the process being measured and the environmental conditions in the area of their installation.

All instruments shall operate in the nominal pressure, temperature, radiation, and humidity environment specified in SDD-93/7. Where the use of smoke detectors is required and the environment extremes can exceed the UL listed normal operating temperature specifications, the fire detection and alarm equipment shall provide monitoring capability and trouble alarms to warn of malfunction or out-of-tolerance conditions.

C. Seismic and Tornado Requirements:

The Plant Fire Protection Instrumentation System is not required to withstand seismic or tornado effects.

7.3.10.1.4 Documentary Criteria

Design codes and standards applicable to the Plant Fire Protection Instrumentation System are given in SDD-93/7.

7.3.10.2 Design Description

The Plant Fire Protection Instrumentation System is described in SDD-93/7 and shown on the plan drawings contained in SDD-26. This system includes fire detection and alarm instrumentation, fire water supply control instrumentation, and instrumentation for control of extinguishing systems.

7.3.10.2.1 Fire Water Supply System Instrumentation

Fire water supply instrumentation is designed to ensure adequate fire water flow capability by automatically starting electrical and diesel driven pumps if water supply loop pressure falls below the nominal operating level. Either pump may be manually operated at local control stations.

7.3.10.2.2 Fire Detection and Alarm Instrumentation

Fire detectors and alarm instrumentation are provided in the containment building, control building, and auxiliary buildings.

Three types of detectors are used: rate compensated thermal, ionization, and infrared. Selection of a sensor type for a particular area is based on the environmental and performance requirements of the area. Each detector provides input to local fire alarm panels.

Signals from the local fire alarm panels (Area Local Control Panel [ALCP] or Fire Control Cabinet [FCC]) initiate alarms in the control room, activate Halon automatic extinguishing equipment, charge preaction extinguishing equipment, and activate space isolation by shutting down local H&V and closing smoke dampers.

Upon loss of normal AC power, the local ALCP/FCC and C-199 detection and alarm circuits are powered by self-contained batteries. The batteries are capable of operating the system under maximum normal load for at least 60 hours and then for five additional minutes in the alarm condition with no external power source (NFPA 72B, 2-9.3.4, 1987 revision). The batteries are maintained at design charge by the normal plant power system.

Inputs are provided to the Plant Fire Annunciator (Panel C-199) for control room alarm in the event of system malfunctions or detection of fires.

7.3.10.2.3 Automatic Extinguishing Equipment Instrumentation

Instruments are provided to measure pressure and monitor valve position in automatic sprinkler systems.

Nitrogen flooding system supply valve position indication and control is provided in the control room. Flooding valves are manually operated at local stations.

The Halon system(s) receive(s) signals from fire detection and alarm instrumentation to automatically reduce air circulation and open Halon flooding valves. All power to the computer can be tripped manually at the operator's discretion.

7.3.10.2.4 Pre-action Extinguishing Equipment Instrumentation

Fire detection and alarm instruments in the control room and cable spreading room provide signals to automatically open charging valves to the control room and cable spreading room sprinkler system. Control switches are provided to permit manual position control of charging valves.

7.3.10.3 Design Verification and Acceptance

Detectors have been chosen for operation in the specific environment of the parameter which they are to measure.

All control and signal conditioning equipment is located in accessible areas available to operators for testing and maintenance. The signal conditioning equipment is designed to withstand the worst case environmental conditions of each location designed for and has been vendor tested to verify this capability. These environmental conditions have been determined from consideration of the building design, heating, and ventilation system and results of various postulated accident conditions. These environmental conditions (40 to 120 °F, 10 to 90% relative humidity) are within the range of normal commercial equipment capabilities in which commercial equipment has been operated for years.

System design has been reviewed by the Associated Factory Mutual Insurance Companies to evaluate adequacy of the design.

An extensive program of preoperational testing has been formulated for the FFTF. This program is designed to verify the proper installation, connection, and operation of the various instrumentation equipment. It will exercise equipment in all modes of operation, verifying that operation is within required specifications under all conditions. This testing program will also demonstrate that no detrimental interactions exist between instrumentation equipment and other plant (electrical) equipment.

7.3.10.4 Design Evaluation

7.3.10.4.1 Level 1

The instruments used were chosen based on performance, reliability, structural integrity and have undergone tests and analysis to verify that they fulfill the Design Criteria set forth in SDD-93/7. The quality of design has been ensured by adherence to the design and quality standards referenced in SDD-93/7.

Redundancy exists in design for protection against component failure to ensure operational availability when analysis shows this to be necessary. Instrument calibrations, alarm setpoints, instrument gain, and sensitivity shall not be disturbed unless directed by plant procedure to ensure operation

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9.8 PLANT FIRE PROTECTION SYSTEM

9.8.1 DESIGN BASES

9.8.1.1 FUNCTIONAL AND PERFORMANCE REQUIREMENTS

- A. The purpose of the Plant Fire Protection System is to establish and maintain for the structures of the FFTF the fire protection required to satisfy an "improved risk" classification in the fire insurance trade.
- B. The containment vessel shall be protected from the effects of sodium leaks and fires so that containment capability is not jeopardized.
- C. All ex-containment secondary sodium areas except the DHX cruciform areas and secondary sodium drain line trench, shall be provided with steel catch pans to prevent contact between liquid sodium and concrete.
- D. The Plant Fire Protection System, in conjunction with the Plant Fire Protection Instrumentation System (SDD-93/7), shall detect and annunciate fires and provide automatic or manually controlled fire extinguishing capabilities. System functions shall include the following:
 - 1. Detection of active or incipient fires
 - 2. Indication of fire location
 - 3. Actuation of local alarms
 - 4. Signaling to the Central Fire Station (refer to SDD-15)
 - 5. Initiation, control, and termination of automatic fire extinguishing systems that supply extinguishment to the fire location
 - 6. Provision for space isolation and manually operated inert gas flooding for sodium fires
 - 7. Actuation of interlocking circuits to shut down fans and/or close dampers in the event of fire
 - 8. Distribution of fire extinguishing agents to potential fire sites.

E. The Plant Fire Protection System shall provide protection against the following classes of fires:

1. Class A: Fires in ordinary combustible materials, such as wood, cloth, paper, and certain plastics
2. Class B: Fires in flammable liquids, gases, and greases
3. Class C: Fires that involve energized electrical equipment
4. Class D: Fires in combustible metals (i.e., sodium and NaK alloy).

F. A municipal fire alarm Radio Transmitter with coded alarm capability shall transmit signals to the Central Fire Station.

G. The Plant Fire Protection System shall use water from the sanitary and fire water loop.

9.8.1.2 DESIGN PARAMETERS

A. The water loop delivery system has the following characteristics:

1. Design flow rate--1,250 gpm
2. Static System pressure--125 psig
3. Water quality--well water strained and chlorinated
4. Primary water supply--252,000-gal reserve in 300,000-gal storage tank T-58, and 250,000 reserve in 325,000-gal storage tank T-330 on site provided by System 26. The reserve is ensured by interior standpipes
5. Backup water supply--300,000-gal storage tank T-87 on site provided by System 23
6. Water is piped separately from each tank to the fire pumps.

B. The Inert Gas Receiving and Processing System (System 82) supplies nitrogen to the N₂ flood patch panels with the following characteristics:

1. Design flow rate--500 scfm
2. System pressure--135 psig
3. Dedicated dewar capacity--3,000 gal liquid N₂.

capacity of this pump, an identical backup pump starts. A secondary sanitary supply is available from T-330 using a similar system with 200-gpm pumps. The loop is also fed by the 1,500-gpm Fire Pumps (electric-driven P-27 and diesel-driven P-28 and P-61), which are automatically activated by low pressure in the loop.

9.8.2.2 AUTOMATIC SPRINKLER SYSTEMS

These systems deliver water to the sprinkler heads installed in designated sprinkler zones (Figure 9.8-2). During normal operation, system piping is filled with water under pressure. When unusual heat, presumably from a fire, fuses the link in any sprinkler head, extinguishing flow begins. Included in each system are the alarm check valve, fusible link sprinkler heads, piping from the riser Butterfly valve to the sprinkler heads, sprinkler alarms, and status monitoring and test connections.

9.8.2.3 PRE-ACTION SPRINKLER SYSTEMS

These systems are used in the Control Room and the Cable Spreading Rooms (Figure 9.8-3). Sprinkler heads and supply piping are provided as in the automatic sprinkler system, but system piping is maintained free of water beyond the special "preaction" valves. The preaction valves are opened by signal from fire detectors in rooms protected by the preaction system and allow the lines to be filled with water. Water is discharged only when the fire is sufficient to melt the fusible link on a sprinkler head. Reset and draining capability is provided to return the system to ready status when the fire is extinguished and the system has been turned off manually.

9.8.2.4 SPACE ISOLATION AND INERT GAS FLOODING SYSTEM

This system provides for extinguishment of sodium reactions by oxygen starvation (Figure 9.8-4). Spaces housing sodium and NaK equipment inside containment are provided with N_2 atmosphere that will limit a potential sodium-oxygen reaction. Control of this inerting system and the equipment necessary to provide the inert atmosphere is provided by interfacing Systems 25, 27, and 82. Most of the spaces containing secondary sodium outside of containment use oxygen depletion with space isolation and manually

operated N₂ gas flooding for extinguishing a sodium fire. The N₂ is supplied from the Inert Gas Receiving and Processing System (System 82). Space isolation is provided by Systems 21 and 25. Space isolation, consisting of H&V shutdown and damper closure, is actuated by the Fire Detection and Alarm System. In addition, a positive drain capability is provided by System 81 to minimize the size of a sodium leak/fire in the secondary HTS pump towers.

9.8.2.5 HALON SYSTEM

The halon system consists of supply containers of Halon 1301, discharging valves controlled by fire detectors, and piped distribution systems configured to each specific space protected. Halon 1301 is used in the Computer Room and the cable space below the Computer Room false floor, room 110, G-3 Gas Turbine enclosure, and FSF Control Room. The activation of the cross-zoned detectors will relay a signal to the Area Local Control Panel (ALCP). The ALCP, in turn, will initiate the halon flood and, at the same time, send a signal to the Control Room Annunciator Panel and the Central Fire Station. After each use, the supply of halon is replenished and the controls are reset for an operationally ready status.

9.8.2.6 AQUEOUS FILM FORMING FOAM SYSTEM (AFFF SYSTEM)

The AFFF System consists of a 1,800-gal water storage tank and a 100-gal storage tank for the foam concentrate. The concentrate control system deposits 6% foam concentrate through a proportional controller into the water stream, where it is distributed to the sprinkler system and is then released as foam in the protected areas. Foam release is through an automatic sprinkler system with quartoid sprinkler heads. Included in this system are the alarms, flow switches, piping, status monitors, and test connections.

9.8.2.7 FIRST AID EXTINGUISHERS

Commercially available manual portable fire extinguishers are distributed throughout the facility as first aid against fires. Three types of extinguishers used are as follows.

1. ABC-type (NH₄H₂PO₄) extinguishers are deployed where Class A, B, or C fires may occur.

2. Halon extinguishers are located near energized electrical equipment for Class A, B, or C fires.
3. Portable dry powder (Na_2CO_3) extinguishers are located where alkali metal fires may occur.

Extinguishers are placed at readily accessible locations along the path of normal personnel traffic. A supply of extinguishers is also provided for special use when maintenance operations with potential fire hazards are performed.

9.8.2.8 FIRE HOSE CONNECTIONS

Hose connections to the wet pipe sprinkler system and meeting NFPA 14 standards are placed at readily accessible locations along the path of normal personnel traffic. Dry, manually connected hose connections are available for fire suppression in containment on the 550-ft elevation and in the stairwell at the 500-ft, 9-in. elevation.

9.8.2.9 FIRE DETECTION AND ALARM SYSTEM

Commercially available fire detectors, alarm devices, and panels are used in the facility in accordance with applicable NFPA codes. Radio master boxes, installed in 1986, comply with NFPA 1221⁽⁸⁾. Fire detection and alarm system equipment upgrades completed in 1992 comply with NFPA 72⁽⁹⁾ and 72E⁽¹⁰⁾. The remaining, originally installed equipment complies with NFPA⁽¹⁾ requirements 72A, 72B, 72C, and 72E, as applicable.

9.8.2.10 MISCELLANEOUS PORTABLE EQUIPMENT

The equipment consists of portable hand-held and wheeled cart-type extinguishers and portable breathing apparatus. This equipment meets the requirements of NFPA 10 and is tabulated in SDD-26, Appendix J.

9.8.3 DESIGN VERIFICATION AND ACCEPTANCE

9.8.3.1 DESIGN ANALYSIS

A. General

1. A stress analysis has been performed to ensure an adequate margin of safety for all anticipated and hypothesized events.
2. A Failure Analysis⁽²⁾ has been performed to ensure system design adequacy.

B. Water Supply System

The design capacity of the fire water system is based on simultaneous requirements for sanitary water (200 gpm), one sprinkler system (800 gpm), and one fire hydrant (250 gpm).

C. Automatic Sprinkler Systems

1. The size of the fire system risers for the automatic sprinkler systems was determined from an analysis of the number of sprinkler zones per building and a flow rate per zone based on NFPA 13⁽¹⁾ for ordinary hazard occupancy (Group 1) at required residual pressure and acceptance flow.
2. Seismic analysis by the static or response spectrum method has been performed to ensure compliance with system seismic criteria.
3. A drainage survey has been performed to ensure that water accumulation from normally operating or ruptured sprinklers would be limited to acceptable areas.

D. Pre-Action Sprinkler Systems

1. The size of the fire system risers to the pre-action sprinkler systems is based on an analysis of the number of sprinkler zones per building and a flow rate per zone based on NFPA 13⁽¹⁾ for ordinary hazard occupancy (Group 1) at required residual pressure and acceptable flow.

2. Seismic analysis by the static or response spectrum method will be performed to ensure compliance with system seismic criteria.
3. A drainage survey was performed to evaluate potential water accumulation from normally operating or failed sprinklers.

E. Halon System

A fire extinguishant study was performed to determine the most effective extinguishant to be used in the Computer Room and the cable space below the Computer Room floor. Halon 1301 was selected.

F. Aqueous Film Forming Foam System

A survey was made of the available extinguishants for a Mobiltherm spill in the lower cooler areas. Aqueous Film Forming Foam was determined to be the most effective.

G. In-Containment Hose Connections

An evaluation of normal manual firefighting capability available for fires not involving alkali metals in containment was performed and a means of providing water in containment for fire suppression was determined to be necessary. Normally dry hose connections to a spare nitrogen containment penetration was determined to be the most effective means of achieving this. The water supply must be connected manually to the ex-containment side of the penetration before water will be available in containment.

9.8.3.2 SYSTEM AND FEATURES TESTS

A. General

1. All fire protection system components are certified by Underwriters Laboratory (UL) and/or the Factory Mutual (FM) System.

If listed or approved components are not available for the intended purpose or use, substitute equivalent items may be used when reviewed and approved by Fire Protection Programs Safety. For each design that invokes this exception, the review process shall include an Unreviewed Safety Question Evaluation.

2. All operable components were shop tested by the vendor or certified operable before delivery to the FFTF site. Show tests will include the control systems whenever possible.
3. All pressure components were pressure tested in accordance with the applicable design codes.
4. All piping and equipment was subjected to nondestructive examination (NDE) in accordance with the applicable specifications, including a hydrostatic test as specified in NFPA 13 and 24.

B. Automatic Sprinkler Systems

Operational tests of all automatic sprinkler systems were performed by the installation subcontractor. Proper system operation was verified by the vendor.

C. Preaction Sprinkler Systems

Operational tests of all preaction sprinkler systems were performed by the installation subcontractor.

D. Aqueous Film Forming Foam System

Operational tests of the AFFF System were performed by the installation subcontractor. Proper system operation was verified by the vendor.

E. Space Isolation and Inert Gas Flooding System

The effectiveness of space isolation and inert gas flooding has been verified by the Large Sodium Fire test program as follows.

1. Test F-1⁽³⁾ demonstrated that a sodium reaction in air could be contained by a metal-lined cell.
2. Test F-2⁽⁴⁾ demonstrated that nitrogen flooding would extinguish a sodium-oxygen reaction.
3. Test F-4⁽⁵⁾ verified that self-extinguishing sumps and cover plates (as used in the secondary pump cells) with nitrogen flooding would limit a large sodium fire.
4. Test F-5⁽⁶⁾ verified that free hydrogen levels will be maintained within nonexplosive limits after a large sodium spill through the use of nitrogen flooding and space isolation.

9.8.4.1.3 AUTOMATIC SPRINKLER SYSTEM

Automatic sprinklers are designed to meet all the requirements of NFPA-13 and will function during fires that generate high local temperatures. The system has been sized with pipe of sufficient diameter to keep full design flow velocities below 10 fps.

The general criteria considered in determining the necessity for automatic sprinkler coverage includes:

1. The potential damage from fire
2. The quantity of combustibles present
3. The extent of ignition sources present, if any
4. The availability of access to the area for manual fire fighting
5. The potential for damage caused by water release.

For example, the cableway in the AEB-West, Area 362, is sprinklered because it has a high concentration of cables, which when ignited may burn with dense black smoke making access impossible. In the two battery rooms, however, there is no apparent ignition source, no high concentration of combustibles, and a fire in one room would not affect safe shutdown of the plant. Therefore, these areas are not sprinklered. Many of the sprinklered areas are provided with ionization detectors for early detection of incipient fires.

9.8.4.1.4 PRE-ACTION SPRINKLER SYSTEM

The pre-action systems meet the same requirements as the automatic sprinkler systems. The pre-action system provides early alarm of an incipient fire through the use of detectors that alarm and automatically open the pre-action valve to fill the pipe. If the system is actuated and discharges into the control room, electrical equipment within the panels would not be subjected to direct spray.

9.8.4.1.5 SPACE ISOLATION AND INERT GAS FLOODING SYSTEM

The principal means used to mitigate the effects of a sodium spill are limitation of spill size and oxygen starvation. All sodium systems contain

isolation or block valves and those that contain large amounts of sodium are capable of being rapidly drained to large storage vessels.

In addition, primary sodium piping within containment is in normally inerted, steel-lined cells. The liners are of two types, the first being a "hot" liner constructed of a steel liner over fire brick. This "hot" liner is designed to contain the maximum design basis sodium spill in that area, thus protecting the underlying concrete from thermal damage during a sodium spill. The "cold" liner, consisting of a steel plate mounted against the concrete, extends from the top of the "hot" liner and covers the remainder of the cell walls (and ceiling) to limit in-leakage of air into the inert space and to limit damage to the concrete from a possible sodium spray. The nitrogen atmosphere purity is monitored and controlled by the Inert Gas Receiving and Processing System (SDD-82).

Spark plug leak detectors are provided on all liquid metal systems to detect sodium leaks. In addition, a sodium leak would be detected by a rise in hydrogen content of the atmosphere caused by water-sodium reaction with the water vapor in the atmosphere.

During maintenance periods, the normally inerted area will be backfilled with air using the inert gas cooler blowers that have been equipped with smoke detection devices. Portable sodium carbonate extinguishers will be provided at such times to be used to combat any resultant reaction. Most of the secondary sodium outside of containment is in cells provided with steel catch pans to hold the maximum credible sodium spill and limit the effect of spilled sodium on the structural concrete. The three secondary sodium pump towers are also provided with a covered, steel-lined sump to reduce the surface area of the potential sodium spill pool and enable optimum blanketing with nitrogen. A positive drain capability is also provided to minimize the size of the sodium leak. Ionization detectors in these areas are interlocked with the fire and smoke dampers to isolate the affected area on detection of a fire.

For the DHX modules, leak detection is either by visual observation of smoke from a DHX module or by a low secondary expansion tank level alarm. In addition, during plant shutdown, periodic DHX module surveillance for detection of small sodium leaks will be accomplished with a portable monitor

Corridors 108, 352, 357, and 304 are designated Seismic Category I to prevent missile damage to Seismic Category I systems in these areas. All other portions of the system are Seismic Category III except supports for the AFFF tanks. The AFFF system has supports designed for Seismic Category I Criteria.

9.8.4.2.9 TORNADO

Portions of the Fire Protection System that are located below the 550-ft level or in the Control Building will not sustain any direct damage from a tornado since the buildings are protected from tornado damage.

9.8.4.3 HYPOTHETICAL ACCIDENT EVALUATION

The Fire Protection System is not required to mitigate the consequences of the HCDA. However, the system would remain operable following an HCDA and capable of performing its automatic function. Manual-actuated activities, such as nitrogen flooding, may be prevented by radiation.

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11.6.1.3 Service and Transient Conditions

- A. The Core Component Conditioning Station (CCCS) design shall provide for the following approximate quantity of new core components to be inerted and preheated for transfer into IDS during each 130-day reactor operating period:

	<u>Quantity</u>
1. Driver Fuel Assemblies	25
2. Control Rod Assemblies	4
3. Reflector	11
4. Core Component Pots	1
Total	41

- B. The transfer cell design shall provide for the following approximate quantity of radioactive containers to be transferred in each 130-day operating period:

Containers of irradiated fuel pins (offsite exam): 11

- C. The TACS design shall provide for three new test assemblies to be inerted, preheated and held, and for three recycled CLIRAS to be held at temperature for subsequent transfer to the reactor. The CLIRA cells can store up to three 12- and/or 40-ft unirradiated assemblies. A 12-ft-long adapter tube would be used to reconfigure the cell to store a 12-ft assembly.
- D. The thermal design of all conditioning stations shall be such that during upset and emergency conditions:
1. Fuel pin cladding shall not exceed 1,500 °F
 2. Concrete shall not exceed 250 °F.
- E. All vaults shall be designed to withstand all normal dead and live loads and setdown loads.
- F. Cells shall be designed to not rupture or release radioactivity in case of a dropped component, test assembly, plug, or handling machine grapple.
- G. Loss of power and cooling shall not lead to a rupture of the pressure retaining boundary or be a hazard to personnel.

- H. Seismic design shall be in accordance with JABE-WADCO-02⁽¹⁹⁾, with the following classifications:
1. Seismic Category: III
 2. Design Code: ASME Section III⁽²⁰⁾ and UBC⁽²¹⁾
 3. Code Classification: Class 3
 4. Load Combination: Dead + Live + Zone 2
 5. Allowable Stress: Per ASME, Section III⁽²⁰⁾
 6. Analytical Method: Static.
- I. The shielding shall be designed to comply with the requirements of Energy Research and Development Administration Manual (ERDAM) Chapter 0524 as interpreted by the guidelines defined in Final Safety Analysis Report (FSAR) Section 11.1, based on the expected system logistics.
- J. The TACS new storage vault shall provide for the following approximate quantity of SERF Cask Transfers -
1. Reactor Characterization Program - 30 transfers
 2. Annual receipt and shipment of pin samples - 15 transfers

11.6.1.4 Design Safety Margins for Hypothetical Events

The Conditioning Stations are not involved in the Hypothetical Core Disruptive Accident (HCDA) and the consequences of the event do not compromise the capability of the stations to protect the stored assemblies.

of 4.5 kW ^(4,5). This provides a design margin to accommodate future test assemblies that may operate at power densities up to 1/3 greater than current FFTF fuel tests. The CLIRA cells can store up to three 12-ft and/or 40-ft unirradiated assemblies. A 12-ft-long adapter tube would be used to reconfigure the cell to store a 12-ft assembly.

11.6.2.1.3.5 Fuel Transfer Port Plug and Utility Cell Storage Vault

The fuel transfer port plug and utility cell storage vault (designated the Shield Plug Storage Station, SPSS-2) is located immediately east of the radioactive storage and recycled CLIRA storage vaults. This 14-ft-deep argon-filled vault contains three separate cells. Two cells will store contaminated fuel transfer port plugs, and one cell is designated for general purpose storage of contaminated components. Separate isolation argon purge capability is provided to the two contaminated plug storage cells and the utility cell.

11.6.2.1.3.6 TACS Equipment Gallery

The TACS equipment gallery is 4 ft deep and is located immediately south and west of the TACS vaults. The equipment gallery contains the process piping, valves instrumentation, blower-motor assembly and the argon-to-nitrogen heat exchanger. An argon closed-loop pressure boundary system is provided for the inerting and preheating of test assemblies in the TACS IP vault. Process piping for separate purge capability to the argon filled vaults, recycled CLIRA cells, and SPSS-2 cells is also located in this equipment gallery. Nitrogen supplied to the bottom of the IP vault and the recycled CLIRA vault is allowed to exit at the top of these vaults into the equipment gallery to maintain a 100 °F average temperature on the operating deck face. This cooling nitrogen exits across the argon-to nitrogen heat exchanger into System 25.

11.6.2.1.4 Shield Plug Storage Station (SPSS-1)

The main shield plug storage station vault is 10 ft deep and is located west of the reactor, just south of the primary sodium pump No. 3 (see Figure 11.1-4). The SPSS-1 argon-filled vault contains 12 standard storage positions for clean shield plugs from core component and test assembly stations within containment, 2 separate cells for the reactor refueling port adapter seal plugs, plus 4 air cells for general purpose storage. A 2 ft deep valve pit area,

underside of the top shield plates. These seals maintain the inert atmosphere and no significant amount of gaseous contamination will occur. Argon purge procedures, to System 82 processing, and the seal system design provide backup protection.⁽⁷⁾

Radiation protection is provided by the thick top shield plates, plugs, and surrounding concrete or heavy steel walls that attenuate radiation from components within, including radiation streaming through the stations from adjacent sources. The CCCS TC liner wall is 1.25 in. thick, embedded in concrete. A 6-in. steel wall north of the TC attenuates IDS sources and a 3-in. steel wall in the equipment gallery south of the cell provides protection during transfer operations. The Recycled CLIRA and Radioactive Storage TACS Vaults are enclosed by a 3-ft-thick concrete wall, which is common with the IEM cell on the north side, and three, 6-in.-thick steel walls on the remaining three sides. Additionally, the south and west walls are increased in thickness at the elevation of the stored activated tests to provide for limited access to the adjacent pipeway. Adequate shielding is provided to limit the dose rate on top of the Conditioning Stations to 0.2 mRem/h and limited access areas adjacent to the sides of the stations to 5 mRem/h^(5,6) design or 2 mRem/h averaged over 1 yr.

Subcritical storage of the test assemblies in the TACS is provided by sufficient geometric spacing to decouple the stored assemblies.⁽¹³⁾ Grid plates are provided in the storage locations of the New and Radioactive Storage Vaults to ensure that the spacing is maintained in the event that a disruptive force occurs, which would tend to modify the storage lattice spacing. However, because the grid spacers have not been seismically qualified, the number of stored, fueled assemblies is administratively controlled. These controls ensure that a criticality cannot occur in either the New or Radioactive Storage Vault. The CCCS services only one fuel assembly at a time in each station and SPSS-1 stores only nonfissile shield plugs; therefore, these stations do not have criticality considerations. No criticality condition will occur in these facilities.

The inert atmosphere of the stations minimizes the possibility of a fire. Cells are normally filled with dry argon and are located in nitrogen-filled vaults.

11.6.4.2 Protective Features Evaluation

11.6.4.2.1 Conditioning Stations

Design features are provided to minimize the effect of potential off-normal occurrences such as dropping of a sealed transfer container or test assembly from the maximum loading height. The design provisions to prevent this have been evaluated. In addition, the Conditioning Stations are designed to withstand loss of power in seismic events. These protective features are discussed in this section.

The CCCS transfer cell is designed to accommodate a drop load energy level of 34,400 ft-lb, which is postulated to occur from the transfer container (370 lb) and grapple (1,500 lb) dropping 20 ft (maximum height) into the TC. The cell was found to be capable of absorbing the energy without rupturing.⁽¹⁰⁾ The drop load energy is absorbed by crushing the honeycomb structure in the bottom of the cell. This item is considered sacrificial in the dropping case and is replaceable.

The TACS IP and recycle CLIRA storage cells are designed to accept the drop load with an energy of 140,000 ft-lb, which is postulated to occur from dropping a test assembly (1,300 lb) and the CLEM grapple (1,500 lb) a distance of 50 ft into the cells. It was found that the TACS cells could absorb the drop load energy without rupturing.⁽¹¹⁾ The energy was absorbed in yielding the upper support liner and by crushing the honeycomb absorber column, both of which are considered sacrificial for the drop load event. The drop load energy from dropping a 12-ft assembly (405 lb), NIH-3 grapple (110 lb), and the storage tube (100 lb) is well below the 140,000 ft-lb design drop load.

Transient analyses of CCCS and TACS following an outage of primary and backup electrical power sources have indicated that the maximum clad temperatures of decaying pins in transfer containers or recycled CLIRA assemblies reach the 1500 °F emergency fuel pin cladding temperature limit^(8,9) but are judged to be safe with no release to occupied areas.

Analyses of off-normal conditions show that the CCCS TC and TACS Recycle CLIRA cells, which have been designated Category III, meet the requirements of seismic Category I.⁽¹⁹⁾ The CLIRA cell adapter tube was analyzed to meet the

requirements of Seismic Category III⁽²⁴⁾. Furthermore, the CLIRA pressure boundary will not incur gross failure under Design Basis Earthquake (DBE) loading and produce a release to the cell. During the DBE the maximum stress in the TACS cell wall is 25,800 psi (allowable is 30,400 psi). Under the same input the CCCS wall would be 20,300 psi with an allowable of 36,000 psi. With the cell plugs in place the cell boundaries will remain intact.^(17,18) Loads under interfacing conditions with the fuel handling machines and equipment are discussed in Section 11.1.3.1.4. These facilities will survive the DBE intact and protect the stored fuel pins and assemblies for all expected equipment interfacing conditions.

11.6.4.2.2 Temporary Inerting Systems

In the event of a failure of the CLEM movable valve limit switches, the CLEM valve could continue traveling down after first contacting the top plate of the inerting cell. The inerting cells are not designed to support the full weight of the CLEM movable valve. To prevent the full weight of the CLEM valve from being applied to the cell, several features have been incorporated into the design of the cells as discussed below.

Based on elevation marks on the CLEM itself, the Movable Valve could move down 2.35 in. below the normal elevation of the top of the Floor Valve at TACS before bottoming out on the mechanical stop pins on CLEM. This would occur only if there were no interlocks/switches or manual stopping of the valve. To accommodate part of this downward movement of the CLEM valve, the top of the inerting cell top plate is located 1-1/2 in. lower than the top of the Floor Valve when located at TACS; thus, the CLEM valve must travel down further before contacting the top of the inerting cell. The remaining downward travel of the CLEM valve after it first contacts the top plate (maximum remaining travel is 0.85 in. max) is compensated for by 8 springs, which can compress more than this distance before reaching their maximum safe load as specified by the spring vendor. Further downward movement of the CLEM Valve beyond this point is prevented by the mechanical stop pins on CLEM. Thus, the full weight of the valve cannot be applied to the cell even if the CLEM limit switches do not actuate. In case of a dropped load, such as a core component pot or driver fuel assembly into a cell, the operation would be stopped until an assessment had been made of any damage to the components. The consequences of

such a drop by CLEM, while highly unlikely, would be constrained by established pressure boundaries. Since the temporary inerting cells are used only for handling of nonirradiated fuel, there is no potential for creating a situation that is hazardous to plant personnel.

When mating with the temporary test assembly inerting cell the CLEM MCV lowers to approximately 0.25 in. from the CLEM mating plate, mounted on the 10-in. vacuum valve, and contacts four posts located at four quadrants around the valve. Limit switches on the CLEM MCV are set to stop the motion of the MCV at this point and unload its weight on the posts (~30,000 lb). The posts are designed so that any two posts will support in excess of 800,000 lb without yielding. The operating procedure requires that the location of the four posts and the shimming of the four posts (match-marked) be visually verified before lowering the CLEM MCV and that an observer be in communication with the CLEM operator during the CLEM MCV lowering operation. If the limit switches failed the CLEM MCV would be stopped by the four posts and the CLEM operator without damage. If the posts were not in place or if the shims were not installed, the CLEM MCV would contact the mating plate located on the 10-in. vacuum valve. This could damage the 10-in. valve with loss of the argon atmosphere around the test assembly but would not damage the inerting cell or the test assembly.

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12.1.2.4 Performance Tests

Acceptance performance tests on equipment related to the Liquid Waste System include the Radioactive Waste System Preoperational and Startup Test. Visual and radiation monitoring checks combined with continuous instrument readings during system operation provide prompt detection of malfunction. Cross-checks between level indicators, alarms, flow rates, and records provide continuous indication of performance. Routine and special tests and calibrations of instruments and alarms ensure their accurate functions.

12.1.2.5 Estimated Releases

The expected concentration and annual collection of radioactive isotopes from normal operation and anticipated operational occurrences in the Liquid Waste System are presented in Table 12.1-2.

12.1.2.6 Release Points

There are no release points of the Radioactive Liquid Waste System to the environment. All radioactive liquid waste is processed and shipped to the Hanford Site 300 Area for disposition. It may then be transferred to the Hanford Site 200 Area.

The 300 Area is an area within the Hanford Site where FFTF liquid wastes are shipped for processing and storage. WHC is responsible for waste management operations in conformance with the applicable DOE regulations and standards.

12.1.3 RADIOACTIVE GASEOUS WASTE SYSTEM

12.1.3.1 Design Bases

The radioactive gaseous waste system comprises two separate gas processing systems, RAPS and CAPS. The CAPS portion of the gaseous waste system

shall be designed to receive the gaseous effluent from cells and equipment that may be substantially contaminated during off-normal events. It will also provide the storage and processing capability for radioactive gas to limit controlled release of radioactive nuclides to as low as practicable levels as specified in Appendix I of 10CFR50. The RAPS portion shall be designed to receive all of the primary cover gas argon contaminated with radioactive noble gases emitted during simultaneous operation of the reactor with 1% failed fuel plus four 4-megawatt closed loops operating with leaking fuel. The system shall be designed such that there is no release of unprocessed primary cover gas to the environment. The rate of release of fission gas radionuclides from failed fuel into the cover gas spaces is given in Appendix I of SDD-82. Section 9.3 describes the details of the systems. The main features of the system are as follows:

12.1.3.2 Design Description

A. Radioactive Argon Processing System

Radioactive effluent from the Reactor and Closed Loop cover gas systems is collected by the RAPS. RAPS has two main sections: (1) collection/storage and (2) cryogenic processing. Reactor cover gas pressure is controlled by drawing cover gas into a subatmospheric collection header and then into the vacuum tank by the RAPS compressors. Filters remove particulate matter before the gases pass through the compressors into the surge and delay tank. Under normal operating conditions the average gas flow rate through the surge and delay tank is approximately 3 scfm, and the residence time provides a delay period of 48 hours. Five baffles located inside the tank ensure that the delay period is not shortened by isotopes traveling directly across the tank. Radioactivity levels are reduced significantly due to radioactive decay of short-lived noble gas isotopes.

The cryogenic processing section includes heat exchangers that use liquid nitrogen cooling to reduce the gas stream temperature. The cold

plantwide release of tritium based on a tritium release from the DHX is calculated as 0.088 Ci/d. The expected release of noble gases, based on operation with 0.1% failed fuel in the reactor core and 0.2% failed fuel in two closed loops assuming leakage from various reactor system, would be .0038 Ci/day.

For the tritium release the internal whole body exposure is calculated based on a breathing rate of 2×10^7 cm³/d and a dose conversion factor of 126 millirems per millicurie of tritium inhaled. The dose conversion factor was developed from the parameters assumed in ICRP Publication 2.⁽⁴⁾ The exposure due to noble gas leakage is based on the semi-infinite cloud approximation with the average gamma energy of the noble gas mixture taken as 0.25 MeV per disintegration. The dose estimates for noble gases are based on the assumptions given in Regulator Guide 1.4.

The results of these calculations for distances within the confines of the Reservation boundary are shown on Figure 12.1-4. These dose estimates include contributions from isotopes of argon, krypton, and xenon. Also compared are the ERDAM 0524 exposure guidelines, the measured guidelines, and the measured background at the Hanford site.⁽¹⁰⁾ The exposure falls well below background for all leakage conditions. At the closest occupied area, the Washington Public Power Supply System (WPPSS) Hanford No. 2 site, the dose rate due to FFTF is less than 10^{-3} mRem per year.

The average exposure to the general public within a 100-mile radius of the FFTF site has also been evaluated. The exposure rates shown in Figure 12.1-4 were extended out to 100 miles. These exposure rates at various distances were then combined with the resident population distribution within a 100-mile radius, as listed in FSAR, Chapter 2. The exposure to the general population is 12 man-rem per year from the reactor when it is operated with 0.1% failed fuel in the reactor core, and 0.2% failed fuel in two closed loops. The estimated population does not include transient construction personnel. In contrast, the exposure to this same population due to background radiation is 70,000 man-rem per year. Only inhalation and direct radiation exposures are considered since exposures via other pathways are insignificant.

The FFTF has been designed to limit the radiation exposure to operating personnel to as low as practicable per DOE Order 5480.1, Chapter XI. Shine is the primary source of radiation exposure to FFTF personnel. The design dose rate for work areas that are staffed full time in the containment and throughout the plant is 0.2 mrem/h. This results in a maximum whole-body dose of 8 mrem/week.

The design maximum leakage of radioactive gases into the head compartment will result in a concentration of less than 0.1 MPC. This results in a whole-body dose rate of <0.25 mrem/h inside the head compartment. Because the H&V air flow is from the Containment Building into the head compartment and out the containment H&V system, there will be no leakage of radioactive gases from the head compartment into the containment. Therefore, there will be no radiation exposure in the containment work areas resulting from leakage of noble gases from the reactor.

12.1.4 RADIOACTIVE SOLID WASTE

The details of the Radioactive Solid Waste System are presented in SDD-24, Radioactive Waste. The following sections include the assessment of the solid waste processing from SDD-24.

12.1.4.1 Design Bases

12.1.4.1.1 Functional and Performance Requirements

- A. The function of the Radioactive Solid Waste System is to collect packaged radioactive solid wastes and provide for interim storage of low-level waste before transfer to the Hanford Site 300 Area for disposal.
- B. The Radioactive Solid Waste System shall be designed to handle the general categories of radioactive waste listed in Table 12.1-9. The solid waste preparation for shipment to the Hanford Site 300 Area shall follow the procedures given in References 5, 6, and 7.
- C. The Radioactive Waste System shall be designed to handle the following types and quantities of waste.

1. Low-Level Solid Waste Containing No Sodium

The Radioactive Waste System shall be designed to collect, store, and transfer nonsodium bearing low-level solid waste from the sources listed in Table 12.1-4. The total low-level solid waste quantity used for design basis is 2,300 ft³/yr with a total weight of 15,000 lb.

2. Sodium-Bearing Low- and High-Level Solid Waste

The quantities and sources of radioactive solid waste containing sodium that shall be accommodated by the Radioactive Waste System are listed in Table 12.1-4. In the future, larger quantities of unreacted sodium-bearing waste are planned to be processed at MASF.

D. The principal radiation protection criteria are identified in Sections 1.5 and 1.10.4 of SDD-24.

E. The solid waste packaging procedure conforms to the Hanford Site waste repository requirements, as defined in WHC-CM-2-14, Hazardous Material Packaging and Shipping⁽²⁸⁾ and in WHC-EP-0063, Hanford Site Solid Waste Acceptance Criteria.⁽²⁹⁾

12.1.4.1.2 Summary Description

The radioactive solid waste system routinely collects low-level solid waste. Low-level solid wastes are temporarily stored or scheduled for immediate transfer to the disposition area.

High-level solid wastes are packaged in a shielded cask and immediately transferred to the disposal area.

12.1.4.1.3 Evaluation

All waste is prepackaged in approved containers designed to provide control of contamination within the facility while handling or storing. All solid wastes will be stored in metal containers at the plant site. Packaging procedures ensure that chemically reactive materials are packaged separately to minimize the possibility of fire. Storage will be confined to controlled or isolated areas to prevent inadvertent exposure of personnel. Use of metal storage containers and storage in protected areas precludes access by rodents, wildlife, and other large animals. Shielding of high-level activity wastes

within the solid waste disposal cask will ensure that exposure of handling personnel to radiation from high-activity wastes is minimized.

12.1.4.2 System Inputs and Expected Volumes

A process flow diagram of the Solid Waste System is presented in Figure 12.1-5. The inputs shown in this figure are given in Table 12.1-4, which lists the volume, weight, and expected activity levels. The waste activity levels are defined in Table 12.1-9.

12.1.4.3 Equipment Description

The Maintenance Support System supplies all tools and equipment needed for FFTF component maintenance, including replacement of equipment that will be classified as high-level radioactive, transuranic, and sodium contaminated waste. The system responsibility includes equipment for transfer, cleaning, disassembly, and disposal of waste items. Such equipment includes casks and handling equipment as described in SDD-41.

12.1.4.4 Packaging

Details of the packaging requirements for solid waste are described in WHC-EP-0063, Hanford Site Solid Waste Acceptance Criteria.

The various waste containers used provide containment of the radioactive contamination and minimize radiation exposure to personnel during temporary storage, handling, shipment, and disposal operations.

12.1.4.4.1 Low-Level Wastes

Low-Level wastes are defined in SDD-24 as those having an exposure rate of less than 200 mrem/hr on the surface of the container. The types of containers used for miscellaneous low-level wastes consist of various sizes of cardboard boxes and metal or fiber drums. Generally, the smaller, low-level wastes are boxed and placed in metal containers for temporary storage until transit for disposal. Large items of solid waste, where the presence of transuranic radionuclides can be safely ruled out, are packaged in heavy wood or concrete boxes.

of the Auxiliary Equipment Building East (AEBE) and the unlined cells of the Reactor Service Building (RSB). The monitors will additionally monitor releases from the CAPS effluent after dilution in the Containment Building exhaust flow. Section 9.4 and SDD-25, "Heating and Ventilation System," present a detailed description of the ventilation system, and the location of the monitor in the exhaust stream is shown in P&ID H-4-11501 of SDD-25. The detector station is housed in the AEBE near the exhaust duct from containment.

The monitoring system incorporates a sample pump that draws an isokinetic sample from the exhaust duct. The monitors consist of beta particulate and noble gas channels and a radioactive iodine integrating sampler.

The monitor readings for each channel during normal operation will result from the combined activity levels contributed by the various releases directed into the ventilation exhaust duct.

The beta particulate and noble gas channels are provided with a high-level alarm and a high-high level alarm. The iodine sampler consists of a cartridge of activated charcoal that will be counted in a laboratory on a weekly schedule to provide an integrated measurement of release of ^{131}I .

High-level alarms will alert the operator to abnormal release rates so that the source of activity can be determined and corrective action initiated. The high-high level alarm automatically stops all air intake to the Control Room.

12.1.5.3.2.2 Reactor Service Building Exhaust Monitors. Two radiation detectors in the exhaust from the Reactor Service Building (RSB) H&V system continuously monitor for abnormal gas and particulate radiation levels. The access areas in the RSB are subject to airborne radioactivity from the Inert Gas Receiving and Processing System, The Radioactive Waste System, and spent fuel handling operations. The location of these monitors is shown in P&ID H-4-51251 of SDD-25.

These monitors provide a high and a high-high level alarm. A high-level alarm will alert the operator of an abnormally high release rate so that

corrective procedures may be initiated. A high-high level alarm automatically stops all air intake to the control room.

12.1.5.3.2.3 Reactor Service Building Unlined Cells Exhaust Monitors.

Exhaust monitors continuously monitor the unlined cells and pipeways for gas activity before dilution by the containment exhaust. The unlined cells include the RAPS and CAPS filter and compressor cells, CAPS vacuum and surge tank cell, CAPS cold box cell, RAPS recycle cover gas tank cell, the sodium removal and cleaning cells, the R/A liquid waste cells and pipeways 233, 222, and 219. The process systems enclosed in these areas potentially contain high levels of radioactive noble gases.

The monitoring system air sampler draws air from the exhaust through a gaseous radioactivity monitor and returns the sampled effluent to the purge line to CAPS. The location of the monitor in the exhaust stream is shown in P&ID H-4-11916 of SDD-25.

The high-level alarm will alert the operator to an abnormally high or increasing gaseous activity level. The high-high level alarm automatically closes the discharge isolation valve from the cells and opens the purge line to CAPS where the gaseous radionuclides are processed.

12.1.5.3.2.4 Heat Transport System Service Building South Main Exhaust Monitors.

The HTS-S building houses auxiliary systems and components that contain radioactive gases. These systems and components represent potential sources for release of gaseous and airborne particulate activity. The argon blower and valve cell exhaust is monitored by the system described in Section 12.1.5.3.2.5. The exhaust of these areas is to the HTS Service Building South main exhaust duct. Section 9.4 describes the ventilation system, and the location of the monitor in the main exhaust stream is shown in P&ID H-4-51262 of SDD-25.

A sample pump draws a gas stream through an airborne particulate channel and gaseous radioactivity monitor.

The radioactivity levels detected at the HTS-S main exhaust monitors may result from the release of gaseous activity into the duct from the argon blower and valve cells and also from leakage from the gas sampling systems contained in HTS-S.

The monitors are each equipped with high-level and high-high level alarms. The high-level alarms will alert the operator to an abnormally high activity level. The high-high level alarm provides a control signal to automatically divert the HTS-South exhaust through the filter system and automatically stop all air intake to the control room.

12.1.5.3.2.5 Argon Blower and Valve Cell Exhaust Monitor. The monitor continuously monitors for noble gas activity in the exhaust from the Argon Blower and Valve Cell and contiguous spaces located in the HTS-South. The FSAR, Section 9.4, describes the ventilation system, and the location of the monitor in the exhaust stream is shown in H-4-51263 of SDD-25.

The monitoring system draws a gas stream through a gaseous radioactivity monitor using a beta scintillation detector. The sample gas returns to the cell.

The monitor provides a high-level and a high-high level alarm. The high-level alarm is set to alert the operator to an abnormally high activity level in the cells. The high-high level alarm will automatically divert cell exhaust to CAPS and initiate closure of the outer containment isolation valves in the argon sampling lines.

12.1.5.3.2.6 Reactor Control Building Monitors. The Reactor Control Building ventilation system is monitored by two systems. One monitors the Control Room air supply for particulate and gaseous radioactivity. This system will alarm, isolate the normal ventilation system, and activate the Life Support System (LSS) when predetermined levels are exceeded. The second system also monitors the Control Room air supply and consists of redundant Seismic Category I monitors for high levels of radioactivity and sodium. Chlorine monitors are installed, but they have been deactivated. Excess sodium or radioactivity automatically isolates the Control Room by closing dampers and deenergizing fans. The second system performs its safety function of isolating the Control Room on electrical power failure.

12.1.5.3.3 Radioactive Argon Processing System Monitors

The inlet and outlet gas to the RAPS is monitored to provide control of the system input rate and alert the operator to high activity in the gas stream routed to CAPS for additional processing. The RAPS is described in the FSAR, Section 9.4, and the location of the monitors in the gas stream is shown in H-4-11056 of SDD-82.

Two gas monitors are provided on the inlet stream for redundancy and diversity. Each detector measures gamma radiation through the wall of the piping. One is a geigermuller (GM) tube and the second, a NaI scintillation crystal. Each monitor has a high and high-high level alarm. The high-level alarm will be set to alert the operator of abnormally high concentration. The RAPS inlet monitor limits the radioactivity input rate to the cryogenic section to 377 Ci/minute by controlling the inlet flow rate. The high-high level alarm alerts the operator at that level.

The RAPS outlet gas is monitored with a NaI crystal detector having high and high-high level alarms. The high-level alarm is set at $2 \times 10^{-4} \mu\text{Ci}/\text{scm}^3$ to alert operators of an abnormally high outlet concentration, and the high-high level alarm is set at $10^{-3} \mu\text{Ci}/\text{scm}^3$.

12.1.5.3.4 Cell Atmosphere Processing Monitors

The inlet and outlet to CAPS is monitored to provide control of the system input and to divert the output back to CAPS when predetermined levels are reached. The CAPS is described in the FSAR, Section 9.3, and the location of the monitors in the gas stream is shown in H-4-11090 in SDD-82.

The CAPS inlet monitor consists of a GM tube detector and has a high and high-high level alarm. The high-high level alarm stops inflow to the cryogenic section and alerts the operator to manually adjust the flow rate to limit the input to less than 76 Ci/minute.

The CAPS outlet is monitored by three monitors before being exhausted to the environs through the combined exhaust system. Two GM detectors are provided for redundancy. Each channel is set to trip at $2 \times 10^{-2} \mu\text{Ci}/\text{cm}^3$ and automatically recycle the exhaust back to the inlet header of the CAPS. A lower setting will be made for the high-level alarm, which will alert the operator to abnormally high activity in the exhaust stream.

The third detector is of the NaI scintillator type suitable for measuring low levels of radioactivity and capable of providing gamma spectral information when used with a laboratory multichannel analyzer to assist in identifying the radioisotopes. The channel also has a control function that prevents the CAPS cryogenic unit from being bypassed when the bypass stream has radiation levels greater than $1.2 \times 10^{-3} \mu\text{Ci}/\text{cm}^3$.

12.1.5.3.4.1 In-Containment CAPS Blower Exhaust Radiation Monitors. The in-containment CAPS blower inlet is monitored for iodine and gaseous radioactivity prior to dilution by the containment exhaust. A signal is provided for use by System 82 to isolate the ICC blowers from the vacuum header when a predetermined level is exceeded.

The detector station cabinet, C357, is located in containment. A sample of the blower inlets is pumped at a fixed rate through the detectors and is returned to the containment H&V exhaust system. The detector assemblies are designed to operate when subjected to the maximum containment pressure.

A high radiation or flow failure or circuit failure comparator provides a control signal to System 82 to close an isolation valve that stops CAPS discharge into the containment exhaust system. The high comparator trip level is set at a level that would result in a concentration at the containment exhaust approximating the controlled area MPC for the anticipated mixture of isotopes.

The sample flows through an iodine monitor and through a noble gas monitor. The iodine monitor incorporates a 95% efficient fixed charcoal filter, and the noble gas detector is a beta-sensitive scintillation counter similar to those used in the other System 96 noble gas monitors.

12.1.5.3.4.2 IEM Cell Iodine Radiation Monitor. The IEM Cell atmosphere is monitored for iodine radioactivity. An alarm is provided in the IEM Cell galleries and the 550-ft level of Containment when a predetermined level is exceeded.

The detector station cabinet C358 is located in Containment. A sample of the IEM Cell atmosphere is pumped at a fixed rate through the detector and is

returned to the Containment exhaust. The detector assembly is designed to operate when subjected to the maximum Containment pressure.

A comparator indicating a high or a flow or a circuit failure provides an alarm signal to the galleries for the operator to manually divert the System 82 CAPS exhaust through the CAPS Cryogenic System. The high comparator trip is set at a level that would result in a concentration at the Containment exhaust approximating the controlled area MPC for iodine-131.

The sample flows through an active iodine monitor that incorporates a 95% efficient fixed charcoal filter, and the detector is a gamma-sensitive scintillation counter similar to those used in the other System 96 RAPS/CAPS detectors.

12.1.5.3.4.3 (Deleted)

12.1.5.3.5 Inert Gas Cell Monitors

Continuous monitoring of the atmosphere in the inert gas cells is performed by means of five in-line gas monitors. Each monitor measures and records the activity from a group of cells by periodically scanning each sample line in the group at the rate of one line every two minutes. Table 12.1-11 lists the cells monitored by each monitor. Each monitor is provided with a high-level alarm comparator that is set to alarm at $10^{-4} \mu\text{Ci}/\text{cm}^3$. The high-level alarm will alert the operator to high activity in the particular cell monitored. Sampling connections are provided to permit the operator to obtain a grab sample or to connect a mobile CAM or portable air sampler described in Sections 12.1.5.3.10 and 12.1.5.3.11 for further verification of the activity levels in the cells. The in-line detectors are of the beta-scintillation type and have a range of 10^{-6} to $10^{-1} \mu\text{Ci}/\text{cm}^3$ of noble gas activity.

These monitors are part of the Atmosphere Sampling and Analysis System, which monitors cell atmospheres for other purposes, including oxygen. A complete description of the sampling system is included in Section 9.3. Section 7.3 describes the radiation monitoring system and design evaluation.

12.1.5.3.6 Interim Examination and Maintenance Cell Monitors

The IEM Cell monitors measure radiation levels within the IEM Cell and IEM Cell annex. They provide the capability to continuously monitor the radiation

level inside the cell and allow monitoring of individual core components in the cell as required. One of the two detectors is capable of being moved, using remote handling equipment. The radiation in the cell consists primarily of beta and gamma radiation from irradiated reactor fuel and radioactive sodium that adheres to the core components.

The monitors have a range up to 10^6 R/h, and a remote display is provided in the IEM Cell operating galleries.

12.1.5.3.7 Closed Loop Ex-Vessel Machine

The CLEM radiation monitor is located at the cold wall outlet of the CLEM and uses a GM tube detector. This monitor also provides display in the control cab and recording on magnetic tape. The FSAR, Section 11.3 describes the CLEM.

12.1.5.3.8 Tritium Monitors

Two tritium monitors continuously monitor for gaseous tritium activity to support the fusion MOTA experiment. One monitor samples the air in the vicinity of the glovebox on the main operating floor of the containment. The other monitor samples the activity levels in the containment combined exhaust ventilation/exhaust flow and is located in AEB-E, near the containment exhaust duct. Each monitoring system incorporates a sample pump and two sets of ion chambers to measure the gaseous tritium activity and subtract the effect of external gamma radiation on the tritium measurement. A high-level alarm will alert the operator locally to abnormal radiation levels so that corrective action can be initiated and the source of activity determined. An alarm will also alert the Control Room if abnormal radiation levels are indicated or power is lost to the instrument. A radiation level signal is transmitted to the Plant Data System for indication and recording.

12.1.5.3.9 Interim Decay Storage Cell Monitor

A well is provided outside the IDS vessel for installation of a temporary neutron monitor. The temporary neutron monitor will be used to measure neutron radiation levels of the IDS to confirm the subcritical arrangement of the fuel stored and verify design calculations. The IDS is described in the FSAR, Section 11.5.

12.1.5.3.10 Mobile Continuous Air Monitors

Mobile continuous air monitors (CAMs) are provided to monitor for airborne radioactive gas and particulate concentrations in occupied work areas, monitor the atmosphere of the inerted gas cells by connection to the inert cell atmosphere sampling connectors described in Section 12.1.5.3.5, and monitor the ventilation exhaust flows when the permanently installed instruments are out of service.

A CAM consists of a pump, a sensor, filters for the particulate and iodine monitors, a countrate meter, a recorder, alarms and associated power supplies. Ten CAMs are provided. Six monitor gross beta-gamma activity of particulates collected on a fixed filter. Two measure the alpha activity of particulates collected on a membrane filter. These monitors utilize an adjustable single channel analyzer to select the alpha energies of interest, 5.11 to 5.17 MeV from ^{239}Pu and ^{240}Pu . Two CAMs measure the beta-gamma from radioactive gases.

Each monitor has a continuous strip chart recorder, visual and audible high-activity alarms, failure (loss-of-signal) alarm and provision for connection to the control room to provide the essential analog information and comparator trip signals when they are used as temporary ventilation exhaust monitors.

12.1.5.3.11 Portable Air Samplers

Portable air samplers sample the atmosphere of the inert gas cells for radioactive gas and particulate activity.

The portable samplers, consisting of a pump, a filter, and a gas sample volume, are attached to the inert gas cell connectors, a sample is taken, and then the filter and gas sample are taken to the counting room for analysis. Two types of air samplers are provided, one for continuous service and one for nonroutine measures.

1. Constant Flow Air Samplers

These mobile samplers are designed for continuous service at a flow rate of 3 cfm. The flow rate through the filter is adjustable from

1.75 to 3.75 cfm. The filter head is sized for a 2 inch diameter HEPA filter. Each sampler is equipped with an elapsed time indicator. Approximately 20 samplers are provided.

2. High Volume Air Samples

These portable samplers are designed to take a sample quickly when there is not sufficient time for taking a routine sample and/or when an accurate determination of the radioactivity is not necessary. The high volume samplers are capable of sampling at a rate of 20 cfm through a 4-in. by 4-in. HEPA filter.

A mobile CAM, described in Section 12.1.5.3.10 is connected if continuous surveillance of cell activity is desired.

The portable air samplers are used on a routine basis to determine leakage of radioactivity in the cells and to sample the cells before planned entry. The samplers verify and supplement data taken by the Nitrogen Atmosphere Sampling and Analysis System described in Sections 9.3 and 12.1.5.3.5. Sample filters are taken to the 300 Area Analytical Laboratory for a detailed analysis.

12.1.5.3.12 Cesium Trap Radiation Monitors

The Cesium Trap located in Cell 499B is periodically monitored for source loading using a gamma ionization chamber installed in a detector well located next to the Cesium Trap. This detector has a range of 0.1 to 10^6 R/h and is monitored using portable instruments during Cesium Trap startup, operation, and determination of full source load. The detector can be replaced in the detector well without entering Cell 499B. No alarm or continuous readout is provided with this detector, and operator attention is minimal. Data recorded are used by Engineering for analysis of Cesium Trap performance and efficiency.

The atmosphere sample from inert gas cells 499A through 499D is routed through a sodium aerosol filter upstream of the inert gas cell monitor described in Section 12.1.5.3.5. Adjacent to this aerosol filter is a GM tube detector that provides early detection of a sodium leak, especially from the Cesium Trap in Cell 499B. The detector also measures the radiation level in the

north area of Cell 497 where it is mounted. The ratemeter is located in cabinet C-514 and actuates the panel annunciator for a failure, alert, or high radiation alarm condition. The alert-level alarm will actuate when a higher than normal radiation level is present in the vicinity of the aerosol filter.

12.1.5.4 Sampling

The objective of the radiological sampling program is to quantitatively measure and record radioactivity in the liquid and airborne effluents to (a) determine compliance with applicable effluent standards, (b) determine the adequacy and effectiveness of effluent treatment and control, (c) achieve environmental protection, and (d) achieve levels of radioactivity that are as low as practicable. This program provides the information required by DOE Order 5484.1, "Environmental Protection, Safety, and Health Protection Information Reporting Requirements."⁽³⁰⁾ Additional details of the WHC Effluent Monitoring Program are contained in Chapter 18 and in WHC-CM-7-5, Environmental Compliance.⁽³¹⁾

12.1.5.4.1 Liquid Effluent Sampling

Radioactive liquid effluents are sampled during normal plant operations and before transfer and disposal to determine the activity and radionuclide composition. The results of the analysis of the sample will be included on

TABLE 12.1-10 — (Cont'd)
INERT GAS PROCESSING MONITORS

CHANNEL NUMBER	CHANNEL RANGE ($\mu\text{Ci}/\text{cm}^3$)	CHANNEL FUNCTION	TYPE MONITOR	DESIGN BACK- GROUND (mR/hr)	DETECTOR TYPE	RATE- METER LOCATION	DETECTOR LOCATION	PRINCIPAL ISOTOPES MONITORED	ALARMS	AUTOMATIC CONTROL FUNCTION
RE-52144	1×10^{-10} to 1×10^{-4}	Radiation monitor in-containment for CAPS outlet gas to the Heating and Ventilation exhaust system	Iodine Sampler with Charcoal Filter	0.2	Gamma scintillation detector	C-357	In Containment Adjacent to In- Containment CAPS & Blowers	I-131	1) Circuit failure alarm 2) High level alarm $1 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$ (1-h collection) 3) High-high level alarm $1 \times 10^{-6} \mu\text{Ci}/\text{cm}^3$ (1-h collection)	High-high alarm or Failure Alarm Isolates In-Containment CAPS System
RE-52143	1×10^{-6} to 1×10^{-1}	Radiation monitor in-containment for CAPS outlet gas to the Heating and Ventilation exhaust system	Noble gas	0.2	Beta scintillation detector	C-357	In Containment Adjacent to In- Containment CAPS & Blowers	Kr-85 Xe-133 Xe-135 Kr-87 Kr-88	1) Circuit failure alarm 2) High level alarm in accordance with Section 3, SDD 82. 3) High-high level alarm 1×10^{-3} $\mu\text{Ci}/\text{cm}^3$	High-high alarm or Failure Alarm Isolates In-Containment CAPS System

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TABLE 12.1-10 — (Cont'd)
INERT GAS PROCESSING MONITORS

CHANNEL NUMBER	CHANNEL RANGE ($\mu\text{Ci}/\text{cm}^3$)	CHANNEL FUNCTION	TYPE MONITOR	DESIGN BACK- GROUND (mR/hr)	DETECTOR TYPE	RATE- METER LOCATION	DETECTOR LOCATION	PRINCIPAL ISOTOPE MONITORED	ALARMS	AUTOMATIC CONTROL FUNCTION
RE-96072	1×10^{-10} to 1×10^{-4}	Radiation monitor IEM Cell Iodine	Iodine Sampler with Charcoal Filter	0.2	Gamma scintillation detector	C-358	In Containment Adjacent to IEM Cell	^{131}I	1) Circuit failure alarm 2) Alert Level $1 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$ (1-h collection) 3) High level alarm $1 \times 10^{-6} \mu\text{Ci}/\text{cm}^3$ (1-h collection)	None
RE-96105	R/h 0.1 to 1×10^4	Radiation monitor Cesium trap Cell 499B	Gamma ionization chamber	0.2	Gamma ionization chamber	None	Cell 499B	^{137}Cs	None	None
RE-83051	mR/h 0.1 to 1×10^4	Radiation monitor for Cesium Trap aerosol filter	Particulate	0.2	GM tube	C-514	Cell 497	^{134}Cs ^{137}Cs	1) Circuit failure alarm 2) Alert level alarm 3) High level alarm	None

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TABLE 12.2-2 (Cont'd)

CHANNEL NUMBER	CHANNEL RANGE (mR/hr)	CHANNEL FUNCTION	CHANNEL LOCATION	RATE-METER LOCATION	DETECTOR TYPE	DESIGN BACKGROUND (mR/hr)	ALARMS	CONTROL FUNCTION
RE-83028-A	1 to 1×10^7	PAM (readout in C177B)	Elec. Pene. ZZE-082	Pico. C177B	Ion. Chamber	0.2	1) Circuit failure alarm 2) High level alarm 3) High-high level alarm	None
RE-83028-B	1 to 1×10^7	PAM (readout in C177B)	Elec. Pene. ZZE-019	Pico. C177B	Ion. Chamber	0.2	1) Circuit failure alarm 2) High level alarm 3) High-high level alarm	None
RE-96086-A	1×10^{-1} to 1×10^4	Area Monitor CR	C177B	C177B	GM Tube	0.2	1) Circuit failure alarm 2) High level alarm	None
RE-96086-B	1×10^{-1} to 1×10^4	Area Monitor CR	C177B	C177B	GM Tube	0.2	1) Circuit failure alarm 2) High level alarm	None

12.3 REFERENCES

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2. Seismic Design Criteria for the Fast Flux Test Facility Site, JABE-WADCO-02, July 1972.
3. A. Court, Analysis of Sealing Effectiveness of FFTF Head Compartment Components, WARD-2171-53, May 1975.
4. International Commission on Radiological Protection, Publication 2 (1959), Pergamon Press, Oxford.
5. J. D. Anderson, Specifications and Standards for the Packaging, Storage and Disposal of Richland Operations Office Solid Wastes, ARH-3032, April 1974.
6. "Safety Standards for the Packaging of Fissile and other Radioactive Materials," ERDA Manual, Chapter 0529, and the RL Manual, Chapter Supplement, May 1975.
7. "Nuclear Criticality Safety," ERDA Manual, Chapter 0530, and the RL Manual, Chapter Supplement.
8. "Effluent and Environmental Reporting," ERDA Manual, Chapter 0513, and the RL Manual, Chapter Supplement.
9. Radiation Protection Procedures, WHAN-M-8.
10. Jack J. Fix, Environmental Surveillance at Hanford for FY-1974, BNWL-1910.
11. Federal Radiation Council, Staff Reports No. 1 (1960), No. 2 (1961), No. 5 (1964) and No. 7 (1965), Background Material for the Development of Radiation Protection Standards, Superintendent of Documents, Washington.
12. "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and Water for Occupational Exposure" Report No. 22, 1963, and "Basic Radiation Protection Criteria" Report No. 39, 1971, National Council on Radiation Protection and Measurements Publications, Washington.

13. R. H. Wilson, and T. H. Essig, Criteria Used to Estimate Radiation Doses Received by Persons Living in the Vicinity of Hanford: Interim Report, BNWL-706, July 1968.
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18. Environmental Impact Statement - Liquid Metal Fast Breeder Reactor Demonstration Plant, WASH-1590, (1535 Vol. 3).
19. Operational Health Physics Procedures, HEDL-TME, 72-20.
20. FFTF Shield Design Criteria, WHAN-IR-49.
21. Portable Radiation Survey Manual, BNWL-MA-62.
22. The Hanford Thermoluminescence Multipurpose Dosimeter, BNWL-SA-3955.
23. Thermoluminescence Personnel Dosimetry at Hanford, BNWL-SA-2793.
24. Deleted
25. Environmental Statement - Fast Flux Test Facility, Richland, Washington, WASH 1510, May 1972.
26. Kebele, T. J., "Estimates of Tritium Distribution in the FFTF," HEDL-TME 74-6.
27. Bahtia, S. L., "Justification for Fuel Handling and Storage at FFTF without a Criticality Alarm System," WHC-SD-SQA-CSA-20303, Westinghouse Hanford Company, Richland, Washington, April 20, 1990.
28. Westinghouse Hanford Company, Hazardous Material Packaging and Shipping, WHC-CM-2-14, Westinghouse Hanford Company, Richland, Washington.
29. Westinghouse Hanford Company, Hanford Site Solid Waste Acceptance Criteria, WHC-EP-0063, Westinghouse Hanford Company, Richland, Washington.
30. U.S. Department of Energy, "Environmental Protection, Safety, and Health Protection Information Reporting Requirements," DOE Order 5484.1.
31. Westinghouse Hanford Company, Environmental Compliance, WHC-CM-7-5, Westinghouse Hanford Company, Richland, Washington.

Two-hour fire partitions and fire doors are provided for large rooms and long cableways to divide the areas into smaller, physically isolated fire areas that delay the spread of the fire, allowing time for the fire suppression system to extinguish the fire. Cable penetrations through these partitions are sealed by an approved flame retardant sealant.

In-containment cable concentrations are covered with a flame-retardant coating to reduce fire hazard.

Fire extinguishers are strategically located throughout the FFTF for use in fighting small fires. Other portable equipment, including self-contained breathing apparatus, gloves, and lifelines, are also located throughout the Plant.

Training of FFTF operating and maintenance personnel in fire fighting techniques will ensure prompt, correct action when a fire is discovered.

The Fire Department will be called to respond to any fire and any danger of loss of control of a situation. This group consists of personnel professionally trained for fire fighting, including situations involving radioactive contamination and sodium fires.

Fire alarm systems are located throughout the FFTF complex. Fire protection systems detect and/or extinguish fires while alerting building occupants by actuating audible and visible devices. The systems also provide a coded signal to the Hanford Fire Department.

In addition to the preceding functions, an annunciator in the Control Room is activated. Fire detectors in areas that are vital to FFTF operating continuity will activate specific area annunciators in the Control Room. Instructions to evacuate areas and zones (or take other actions), in addition to evacuating the immediate vicinity of the local alarm, can then be transmitted over the public address system. A description of the FFTF Fire Protection System is contained in the System Design Description 26⁽¹⁷⁾ and the FSAR, Section 9.8.

13.3.6.3.2 Repair and Damage Control

The entire FFTF staff is composed of technically and vocationally trained personnel capable of improvising necessary repair and control measures for correction of an emergency situation. Location of the Plant on the Hanford Site provides a large pool of specialized equipment and outside personnel for additional assistance. Wherever possible, corrective measures are anticipated and included in emergency and operating procedures.

13.3.6.3.3 System Control

System design is aimed at automatic corrective actions, such as plant shutdown, maintenance of safe shutdown, and system isolation, whenever operating parameters become abnormal. Operating procedures are written for manual control of these same situations should automatic features fail.

13.3.6.4 Protective Actions

Protective actions are defined as those measures taken after an accident or uncontrolled release has occurred, for the purpose of preventing or minimizing exposure to persons, likely to occur if the action were not taken. This section describes those protective actions considered, action levels or other criteria for application, and area or population-at-risk involved.

13.3.6.4.1 Protective Cover, Evacuation, and Personnel Accountability

Plans for these actions are categorized for onsite FFTF and offsite FFTF.

A. Onsite FFTF

1. Action Criteria

Plant instrumentation is the basis used for initiating site protective actions. In addition to normal operation functions, the plant Instrumentation and Control Systems provide means to cope

FFTF TECHNICAL SPECIFICATIONS

CHAPTER 17.0

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17.1 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications. Unless otherwise specified, the definition of terms used in the specifications shall be as defined in ANSI N1.1 (latest issue), "Glossary of Terms in Nuclear Science and Technology."

17.1.1 Deleted.

17.1.2 CHANNEL CALIBRATION

The adjustment of the channel output is such that it responds with specified range and accuracy to values of the monitored parameter, which are determined through use of reference instruments that have been related to known standards.

The CHANNEL CALIBRATION shall encompass the entire channel, including the sensor and alarm and/or trip functions and shall include the CHANNEL FUNCTIONAL TEST.

The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

17.1 DEFINITIONS

17.1.3 CHANNEL CHECK

The qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication with other indications derived from independent instrument channels measuring the same parameter.

17.1.4 CHANNEL FUNCTIONAL TEST

The injection of a simulated signal into the channel, as close to the primary sensor as practicable, to verify OPERABILITY, including alarm and/or trip functions.

17.1.5 CHANNEL TIME CONSTANT (Reactor Shutdown System)

The RSS CHANNEL TIME CONSTANT characterizes the time interval from when a step change greater than the trip point is made to the monitored parameter at the output of the channel sensor until the appropriate comparator trips. Included in this TIME CONSTANT are the delays associated with the tripping of the comparator and the delays associated with permissive relay actuation.

17.1.6 CLOSED SYSTEM

A system that penetrates the containment boundary but that does not communicate with the containment atmosphere or the primary HTS system. The inner boundary is seismically qualified and a containment isolation valve is provided excontainment.

17.1.7 CONTAINMENT INTEGRITY

The containment must be isolable by the Containment Isolation System; i.e., actuation of the containment isolation valves by the CIS will effect a leak-tight Reactor Containment Building (RCB) that could be expected to have leakage no greater than the design leak rate.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.1 SAFETY LIMITS

LIMITS

17.2.1.1 REACTOR CORE

Reactor power and flow shall be controlled such that calculated driver fuel cladding strain is maintained below 0.7%.

17.2.1.2 REACTOR HEAT TRANSPORT SYSTEM

The steady-state and transient temperatures and pressures of the HTS shall be controlled so that the heat transport system component boundaries, the reactor vessel, the core support structure, the reactor head, and the instrument tree do not exceed emergency stress allowables or the allowable cumulative damage fractions of the applicable ASME Code.

17.2.1.3 CONTAINMENT BUILDING

The condition of the containment building shall be controlled to ensure that the containment building allowable leakage rate (≤ 0.5 percent per 24 hours of the original weight of the containment atmosphere at 2 psig) is not exceeded whenever containment isolation is required.

17.2.1.4 FUEL STORAGE FACILITY (FSF)

The temperatures and pressures of the FSF sodium system shall be controlled so that the FSF storage vessel does not exceed emergency stress allowables or the allowable cumulative damage fractions of the applicable ASME Code.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.1 REACTOR SHUTDOWN SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

LIMITS

- a. The REACTOR SHUTDOWN SYSTEM shall be OPERATING, and actuation limits and TIME CONSTANTS for each protective function shall comply with the limits shown in Table 17.2.2-1, unless the channel is tripped.
- b. The Primary REACTOR SHUTDOWN SYSTEM Reactor Vessel Level Subsystem, protective logic, and final actuation logic shall be OPERATING, and actuation limits and TIME CONSTANTS for this protective function shall comply with the limits shown in Table 17.2.2-1, unless the channel is tripped.

APPLICABILITY

- a. MODES 1, 2, and 3 (during single control rod testing per Specification 17.3.1.6.c).
- b. MODES 3 and 4, when primary MHTS pumps are operating on main motors.

ACTION

- a&b Any channel not meeting its actuation limits or TIME CONSTANT shall be placed in the tripped condition.

TABLE 17.2.2-1 (Cont'd)

FUNCTIONAL UNIT	REACTOR SHUTDOWN SYSTEM ACTUATION LIMITS		
	SYSTEM*	ACTUATION LIMIT	TIME CONSTANTS
11. Experimenter Trip			
(A) PIOTA Temperature	(P) ^(g)	As specified for each test ^(g)	<0.5 second
(B) Secondary Linear - Nuclear Power Level	(S) ^(g,h)	As specified for each test ^(g)	<0.015 second
(C) Primary-Power Level	(P) ^(a,g)	As specified for each test ^(g)	<0.015 second

*FOOTNOTES:

(P) is Primary Shutdown System; (S) is Secondary Shutdown System.

(a) Bypass permitted when primary wide-range linear power channel indicates >26% FULL POWER

(b) Bypass permitted when primary wide-range log power channel indicates $>1.5 \times 10^{-2}\%$ FULL POWER (60 kW) and pressure permissive (see Table 17.2.2-1, Item 3C) is satisfied.

(c) Bypass permitted when <5% power.

(d) Bypass permitted when flow in corresponding primary loop is <20%.

(e) Automatically instates low primary loop flow and loss of offsite power trip functions at 5% FULL POWER.

(f) Bypass permitted in MODES 3 and 4 when Primary MHTS pumps are not running on main motors.

(g) May be bypassed or adjusted out of the operating range except when required for special reactor testing. When applicable, the ACTUATION LIMIT and the plant conditions defining the applicability will be provided in a Technical Specification waiver.

(h) May be bypassed during power ascent when the reactor power exceeds the Low Power Permissive setpoint, as specified when the function is active. Additionally, the function is automatically bypassed whenever all three primary HTS loops have flow in excess of 20% FULL FLOW.

TABLE 17.2.2-1 (Cont'd)
 REACTOR SHUTDOWN SYSTEM ACTUATION LIMITS
 ACTUATION LIMIT EQUATIONS

EQUATION (1) - Flux ² /Pressure Trip	EQUATION (3) - Flux/Reactor Flow
$P \left[\frac{4}{G/P_F} \right] + \left[1.04 + \frac{25}{G} \right] \leq K \left[\frac{(0.032 F - 0.04)^2}{4} + X - 0.05 \right]$	% FULL FLOW - 0.8705 (% FULL POWER) ≥ -10.135

EQUATION (2) - Pressure Permissive (above 60 kW)

$$P \frac{4}{G/P_F} + 1.04 + \frac{25}{G} \leq K \quad 1.538 + X$$

X, K, P_F, G, P, and F are defined as follows:

$$X = 0.8067 \left(\frac{0.96G - 25}{P_F} \right) - 0.060 \text{ volts}$$

$$K = \frac{0.96 - \frac{25}{G}}{X + 0.060}$$

P_F = FULL FLOW pressure in psig*

G = full scale pressure instrument channel output in psig* (G = 1.25 P_F)

P = fraction of FULL FLOW pressure

F = percent FULL POWER

*Measured with respect to containment pressure at a specified tap.
 Equation (2) is derived empirically to use gauge pressure.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.2 REACTOR HEAT TRANSPORT SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

LIMITS

Each component listed in Table 17.2.2-2 shall have at least one relief path capable of providing pressure relief at no greater than the listed actuation limit.

APPLICABILITY: ALL MODES.

ACTION

MODES 1 and 2:

- With no relief path OPERABLE for the primary system, initiate a REACTOR SHUTDOWN and subsequent to the REACTOR SHUTDOWN transfer the primary HTS pumps to pony motors, unless also responding per the Action of Section 17.3.2.10, until an alternative relief path is established.
- With no relief path OPERABLE for a secondary loop, initiate a REACTOR SHUTDOWN and subsequent to the REACTOR SHUTDOWN transfer the secondary HTS pump to pony motor until an alternative relief path is established.

MODES 3 and 4:

- With no relief path OPERABLE for the primary system, terminate in-vessel component handling operations, or if the primary HTS pumps are operating, transfer to pony motors, unless also responding per the Action of Section 17.3.2.10, until an alternative relief path is established.
- With no relief path OPERABLE in a secondary loop, transfer the secondary HTS pump to pony motor until an alternative path is established.

TABLE 17.2.2-2

HTS PRESSURE RELIEF VALVE ACTUATION LIMITS

REGION	COMPONENT	RELIEF VALVE	ACTUATION LIMIT (PRESSURE PSIG)
<u>Reactor Heat Transport</u>			
Low Pressure Primary	T-42	52042	16.5
Low Pressure Primary	T-42	52043	16.5
Secondary HTS Loop 1	T-7	52754	132
Secondary HTS Loop 1	T-7	52756	132
Secondary HTS Loop 2	T-8	52631	132
Secondary HTS Loop 2	T-8	52632	132
Secondary HTS Loop 3	T-9	52606	132
Secondary HTS Loop 3	T-9	52607	132

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.3 CONTAINMENT BUILDING - CONTAINMENT ISOLATION SYSTEM ACTUATION LIMITS

- a. The Containment Isolation System (CIS) shall be OPERATING and actuation limits and RESPONSE TIMES for each protective function shall not exceed the values shown in Table 17.2.2-3, unless the channel is tripped.
- b. At least one vacuum relief path shall be provided for the containment building with a relief setting no more negative than -18.1" water gauge.
- c. The actuation pressure for at least one pressure actuated Containment Isolation Valve (listed in Table 17.3.4-1) on each penetration shall be 10 psig or greater. Any penetration with pressure actuated Containment Isolation valves shall be isolated with at least one Containment Isolation Valve if either isolation valve does not meet this actuation limit.

APPLICABILITY

- a. When CONTAINMENT INTEGRITY is required per Specification 17.3.4.1. Additionally, the H&V portion of the CIS shall be OPERABLE when the ICCB blowers are exhausting to containment H&V.
- b. All MODES
- c. When CONTAINMENT INTEGRITY is required per Specification 17.3.4.1.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.3 CONTAINMENT BUILDING - CONTAINMENT ISOLATION SYSTEM ACTUATION LIMITS

LIMITS

ACTION

- a. Any channel not meeting its actuation limit RESPONSE TIME shall be placed in the tripped condition.
- b. If no vacuum relief path is OPERABLE, the plant shall be placed in a condition not requiring CONTAINMENT INTEGRITY and then an alternative vacuum relief path shall be established.
- c. The affected containment penetration shall be isolated with at least one Containment Isolation valve, or the plant shall be placed in a condition not requiring CONTAINMENT INTEGRITY.

17.2.2.3 CONTAINMENT BUILDING - CONTAINMENT ISOLATION SYSTEM ACTUATION LIMITS

TABLE 17.2.2-3

CONTAINMENT ISOLATION SYSTEM ACTUATION LIMITS

<u>FUNCTIONAL UNIT</u>	<u>ACTUATION LIMITS</u>
1. <u>CONTAINMENT ISOLATION</u>	
a. Principal Containment	
(1) Manual	OPERABLE
(2) Containment Exhaust Radiation High Trip Point	1×10^3 mR/hr
(3) Head Compartment Exhaust Radiation High Trip Point	1×10^3 mR/hr
b. H&V Containment	
(1) Manual	OPERABLE
(2) Containment Exhaust Radiation Low Trip Point	2.3 mR/hr
(3) Head Compartment Exhaust Radiation Low Trip Point	8.5 mR/hr
2. <u>RESPONSE TIMES (CIS)</u>	
a. Instrument channel RESPONSE TIMES shall not exceed 3.0 seconds for any incident radiation level in excess of 20 mR/hr.	
b. The H&V exhaust valves (HV-87003 and HV-87004) closure times shall not exceed 3.0 seconds.	

17.3/4.1 REACTOR

17.3/4.1.2 CORE LOADING CHARACTERISTICS

LIMITS

17.3.1.2 LIMITING CONDITIONS

The core loading characteristics shall be established as follows:

- a. The irradiation lifetimes of driver fuel pins shall not exceed the values shown on Figure 17.3.1.2-1.
- b. The allowable lifetimes for the absorber assemblies shall not exceed:
 - b.1 Series 1.5 and Series 2 Absorber Assemblies:
 - Row 3 Primary Rods-- 15.0×10^{22} n/cm² (E > 0.1 MeV) axial average fast neutron fluence delivered to the peak duct corner.*
 - Row 5 Secondary Rods-- 15.0×10^{22} n/cm² (E > 0.1 MeV) axial average fast neutron fluence delivered to the peak duct corner.*
 - b.2 The allowable accumulated exposure for Peripheral Shim Absorber Assemblies shall not exceed 420 EFPD at 400 MW.

*Requires rotation per following prescribed schedule. Without rotation, the fluence limit is 11.8×10^{22} n/cm².

If first rotation Performed at:	Then Second Rotation to be Performed at:
$5.0 \leq \phi t < 7.0$	$8.8 \leq \phi t \leq 10.8$
$7.0 \leq \phi t < 9.3$	$10.8 \leq \phi t \leq 12.8$
$9.3 \leq \phi t \leq 11.8$	Not Required

17.3.1.2 CORE LOADING CHARACTERISTICS

*Power is assumed to decrease linearly by 15.5% at a peak burnup of 80,000 MWd/MTM.

FIGURE 17.3.1.2-1. Steady-State Fuel Pin Lifetimes.

17.3/4.1 REACTOR

17.3/4.1.2 CORE LOADING CHARACTERISTICS

LIMITS

17.3.1.2 LIMITING CONDITIONS (Cont'd)

- c. The reactor shall not be operated unless the following conditions are met:
 - c.1 No assembly located in a DFA position shall operate at an outlet temperature greater than 210 °F above the average outlet temperature of the six adjacent assemblies, and
 - c.2 No assembly (including OTAs) shall operate at an outlet temperature more than 270 °F above an adjacent assembly.

APPLICABILITY

a-c MODES 1 and 2.

ACTION

- a&b If the conditions are not met, perform a REACTOR SHUTDOWN.
- c. If the conditions cannot be met (by analysis), REACTOR STARTUP shall not be initiated. If, during reactor operation, the previous analysis is judged to be incorrect and the limits are not being met (either by reanalysis or review of operating data by the engineering organization), a REACTOR SHUTDOWN shall be initiated to restore temperatures to within the specified limits and a SAFETY REVIEW shall be performed.

17.3/4.1 REACTOR

17.3/4.1.2 CORE LOADING CHARACTERISTICS

LIMITS

17.4.1.2 SURVEILLANCE REQUIREMENTS

a&b The irradiation history of all core assemblies shall be evaluated and documented for each CYCLE before the initial REACTOR STARTUP to determine the acceptability of loading all core positions, to ensure that these conditions will be satisfied throughout the CYCLE. Fuel pin lifetimes may be obtained either by using Figure 17.3.1.2-1 or by performing an explicit calculation, using the method of Reference 55 (which was used to prepare the figure) and accounting for actual conditions during previous operation. A detailed description of the procedure used to implement the method developed in Reference 55 for explicitly evaluating driver fuel lifetimes is included in Reference 98.

For those fuel pins whose continued irradiation past the limits of the figure is justified by individual analysis, it is necessary to also evaluate the continued irradiation of the associated duct as part of the Reload Design Report documentation.

c. Each core loading shall be evaluated following any change in core loading before a REACTOR STARTUP to establish that the calculated assembly outlet temperatures are within the specified limits. Surveillance and evaluation of the operating data during reactor operation are the responsibility of the engineering organization. This documented technical review shall be performed at the beginning of each CYCLE; it shall be performed again at mid-CYCLE for CYCLES expected to exceed 50 EFPD.

17.3/4.3 INSTRUMENTATION

17.3/4.3.1 REACTOR SHUTDOWN SYSTEM INSTRUMENTATION

LIMIT

17.3.3.1 LIMITING CONDITIONS

The limits for this Type "a" specification are stated in Specification 17.2.2.1.

APPLICABILITY

See Specification 17.2.2.1.

ACTION

See Specification 17.2.2.1.

17.4.3.1 SURVEILLANCE REQUIREMENTS

- 1) Each Reactor Shutdown System instrumentation channel shall be demonstrated to be OPERATING by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 17.4.3-1.
- 2) Checks of the ex-vessel neutron detector channels, based on a calorimetric calculation of reactor power (and adjustment if required), shall be performed during each REACTOR STARTUP and daily when at $\geq 10\%$ reactor FULL POWER. (When transitioning from a SHIMMED CONFIGURATION, see Specification 17.3.1.13.b.4.)

17.3/4.3 INSTRUMENTATION

17.3/4.3.1 REACTOR SHUTDOWN SYSTEM INSTRUMENTATION

LIMITS

17.4.3.1 SURVEILLANCE REQUIREMENTS (Cont'd)

- 3) The instated bypass permissive shall be demonstrated to be capable of automatic removal during the at-power CHANNEL FUNCTIONAL TEST of the channels affected by bypass operations.
- 4) The Reactor Shutdown System CHANNEL TIME CONSTANTS, LOGIC DELAY TIME and SCRAM RELEASE TIME shall be demonstrated to be within their limits stated in Section 17.2.2.1 at the intervals listed in Table 17.4.3-1, and following replacement of components affecting the dynamic response.
- 5) Checks of the primary and secondary permanent magnet flowmeter channels (and adjustments, if required) shall be performed during the initial REACTOR STARTUP of each CYCLE and monthly thereafter during reactor operation.
- 6) Checks of the primary loop pressure channels (and adjustments, if required) shall be performed during each REACTOR STARTUP following refueling (MODE 4) operations (these checks may be omitted if allowed by a prior SAFETY REVIEW) and monthly thereafter during reactor operation.

17.3/4.3 INSTRUMENTATION

17.3/4.3.2 CONTAINMENT ISOLATION SYSTEM INSTRUMENTATION

LIMITS

17.3.3.2 LIMITING CONDITIONS

The limits for this Type "a" specification are stated in Specification 17.2.2.3.a.

APPLICABILITY

See Specification 17.2.2.3.a.

ACTION

See Specification 17.2.2.3.a.

17.4.3.2 SURVEILLANCE REQUIREMENTS

Each CIS instrumentation channel shall be demonstrated OPERATING by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 17.4.3-2.

The CIS RESPONSE TIME of each CIS function shall be VERIFIED at least once each year to be within the limit stated in Section 17.2.2.3 by insertion of the detector check source.

17.3/4.3.2 CONTAINMENT ISOLATION SYSTEM INSTRUMENTATION

TABLE 17.4.3-2

CONTAINMENT ISOLATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>
<u>CONTAINMENT ISOLATION</u>		
1. Manual	N.A.	R
2. Containment Exhaust Radiation	S	M*
3. Head Compartment Exhaust Radiation	S	M*
4. Actuation Logic	N.A.	M
5. Breakers	N.A.	M

NOTE: CHANNEL CALIBRATION is to be performed at least once each year.
*Except for G-M tubes that are checked weekly.

Frequency Notation

S	MONITOR
M	At least once every 30 days
N.A.	Not applicable
R	At intervals not to exceed 120 days

17.3/4.3 INSTRUMENTATION

17.3/4.3.5 INTERLOCKS INSTRUMENTATION

LIMITS

17.3.3.5 LIMITING CONDITIONS

The reactor shall not be operated (item a) or OPERATED with the rods in automatic (items b & c) unless the following interlocks are OPERABLE:

- a. Secondary rod withdrawal is inhibited if more than one primary rod is not fully withdrawn (except for LOW POWER PHYSICS TESTS).
- b. If more than one secondary rod is selected for automatic control, automatic action by any rod is inhibited; and
- c. Automatic out motion of a secondary rod cannot be accomplished unless all primary rods are fully withdrawn (except for LOW POWER PHYSICS TESTS).

APPLICABILITY: MODES 1 and 2 (except for operation in the SHIMMED CONFIGURATION as controlled by Specification 17.3/4.1.13.).

ACTION

- a. Initiate a REACTOR SHUTDOWN and restore the INOPERABLE interlocks to OPERABLE status.

b&c Place rod control in manual

17.4.3.5 SURVEILLANCE REQUIREMENTS

Each of the above interlocks shall be determined OPERABLE before criticality for each REACTOR STARTUP after a planned refueling shutdown, resetting the ARPI interlocks from the SHIMMED HEIGHT to 36 in. in MODE 3 or 4 (see Specification 17.3.1.13.b, Bases Applicability) or after interlock equipment maintenance.

17.3/4.4 CONTAINMENT SYSTEM

17.3/4.4.1 CONTAINMENT INTEGRITY

LIMITS

17.3.4.1 LIMITING CONDITIONS

- a. The reactor shall not be operated or be in the conditions specified in any of the Applicability "a" items below unless CONTAINMENT INTEGRITY is maintained.
- b. Containment atmosphere pressure shall be maintained between -18.1 in. H₂O (-0.656 psig) and +10.0 psig (except for local leak rate testing for which maximum pressure shall not exceed 11 psig). Containment structural temperature shall be maintained below 250 °F. If planned activities on the active containment boundary or on inactive portions that will be reactivated before retest may result in local conditions in excess of these limits, such activities shall be subjected to a prior SAFETY REVIEW.

APPLICABILITY

- a.1 Whenever in MODES 1, 2, or 4.
- a.2 Whenever in MODE 3 for the first 24 h following a REACTOR SHUTDOWN or scram (unless decay heat falls below 400 kW).
- a.3 Whenever airborne activity at the containment building exhaust exceeds 2.3 mrem/h.
- a.4 Whenever the iodine activity contained in pins and assemblies in the IEM cell exceeds 50,000 Ci. (See Section 17.3.7.2.3.d.)
- a.5 Whenever the BLTC is transferring an assembly or fuel container from the IDS. (CONTAINMENT INTEGRITY is required until the assembly or fuel container has been VERIFIED to be acceptable for transfer outside containment. See Section 17.4.7.2.6, "Surveillance," Part a.)
- a.6 Whenever Specification 17.3.2.3.b.4 is not met.
- b. All MODES.

17.3/4.4 CONTAINMENT SYSTEM

17.3/4.4.2 CONTAINMENT ISOLATION VALVES

LIMITS

ACTION

- a. With one or more of the isolation valve(s) specified in Table 17.3.4-1 INOPERABLE, either isolate each affected penetration by closure of at least one containment isolation valve as soon as possible or place the plant in a condition for which CONTAINMENT INTEGRITY is not required. (See also "Safety Actuation Limit," Section 17.2.2.3.b and c.) Valves in closed systems are exempt from this but must comply with Specification 17.3/4.4.3.
- b. With a normally closed isolation valve or other containment penetration closure open or unsecured, isolate the penetration with at least one secured containment isolation valve as soon as possible or place the plant in a condition not requiring CONTAINMENT INTEGRITY.
- c. With an auxiliary valve not in the position specified in Table 17.3.4-1, either return the valve to the proper position, isolate the penetration by closure of at least one containment isolation valve, or place the plant in a condition not requiring CONTAINMENT INTEGRITY.
- d. With any IEM Cell Sodium Removal System CIV INOPERABLE, at least one valve for the affected penetration shall be placed in the closed position or the plant shall be placed in a condition not requiring CONTAINMENT INTEGRITY. If a fueled assembly is being gas-cooled by the Sodium Removal System in the downflow cooling mode or is being processed in the cleaning station, it shall be restored to a configuration that can be cooled without having SRS CIVs open before the penetration is isolated. After 12 hours, the penetration shall be isolated or the plant placed in a condition not requiring CONTAINMENT INTEGRITY.

17.3/4.4 CONTAINMENT SYSTEM

17.3/4.4.2 CONTAINMENT ISOLATION VALVES

LIMITS

17.4.4.2 SURVEILLANCE REQUIREMENTS

- a. The isolation valves specified in Table 17.3.4-1 shall be VERIFIED OPERABLE as follows:
 - a.1 VERIFY at least once every 30 days that, upon the H&V Isolation Test Signal, each isolation valve that is designed to respond to that signal actuates to its isolation position. At least once each year the time required for closure of each H&V isolation valve shall be measured and verified to be ≤ 3 seconds.
 - a.2 VERIFY at least once every 30 days, upon the principal isolation test signal, each isolation valve that is designed to respond to that signal, actuates to its isolation position.
 - a.3 At least once each year, each normally open containment isolation valve that is actuated by electrical signals (other than H&V or principal containment isolation) shall be operated through at least one complete cycle of full travel.
 - a.4 The pressure actuated CIS valves shall be actuated once each 125 days or once each CYCLE, whichever is the longer period. Annually calibrate each pressure actuated CIS valve setpoint. Upon actuation, each valve designed to actuate upon loss of system pressure (LOSP) must close.
 - a.5 A cycling test shall be performed immediately prior to returning any valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit.

17.3/4.5.5 COVER GAS ACTIVITY

FIGURE 17.3.5.5-1. Maximum Allowable Reactor Cover Gas Activity Versus Reactor Head-Mounted Component Purge Flow (Based on 1% Failed Fuel).

17.3/4.5 PLANT SYSTEMS

17.3/4.5.6 PRESSURE RELIEF PATHS

LIMITS

17.3.5.6 LIMITING CONDITIONS

The reactor shall not be operated unless the minimum required pressure relief paths listed in Table 17.3.5-2 are OPERABLE or an alternate equivalent path is provided (subject to a SAFETY REVIEW). (See also, Safety System Actuation Limits 17.2.2.2 and 17.2.2.3.b.)

APPLICABILITY: All MODES.

ACTION

MODES 1 & 2:

- With less than the minimum number of relief paths OPERABLE for the primary system initiate a REACTOR SHUTDOWN and establish an alternate relief path.
- With less than the minimum number of relief paths OPERABLE for the secondary system, a REACTOR SHUTDOWN shall be initiated and subsequent to the REACTOR SHUTDOWN the secondary HTS pump transferred to pony motor until an alternate relief path is established.
- With less than the minimum number of relief paths OPERABLE for the containment vacuum relief system, place the plant in a condition not requiring containment integrity and maintain it in this condition until an alternate relief path is established.

17.5 DESIGN FEATURES

17.5.3 REACTOR CORE

17.5.3.2 CONTROL ROD ASSEMBLIES AND DRIVE MECHANISMS

The reference safety and control rods contain boron carbide. In addition to reference absorbers, non-reference absorbers satisfying Technical Specification 17.3.9.3 may also be irradiated in Row 5 positions. The following design features are applicable to the reference absorber assemblies and to all drive mechanisms. Any modification to the reference control rod subsystem, which includes the absorber assemblies and associated driving mechanisms, must not change these design features:

- a. The primary rods are located in Row 3 and the secondary rods are located in Row 5 of the reactor core and contain natural boron carbide at 92% of nominal theoretical density. When the rods are raised to the upper mechanical limit of travel, the boron carbide is out of the core and the maximum increase in reactivity is achieved.
- b. Control rod drive mechanisms are provided for each primary or secondary rod to control the vertical position of the rods. The control rod drive mechanisms can raise or lower each safety or control rod individually at a speed not exceeding 9.8 in./s over a distance of 37 in.
- c. In the event of a scram, the control rod drive mechanisms release the driveline and provide spring assist during the top 14 in. of travel. Control rod drive mechanisms provide, with plug-in instrumentation, the capability for timing scrams.
- d. The control rod drive mechanisms are energized from motor-generator sets to provide ride-through capability for momentary electrical power outages of less than 24 cycles. The primary control rod mechanism motor-generator set is supplied from the Main Heat Transport System Power System Loop No. 1 bus to provide inherent rod release in event of loss of power to the prime movers of the main coolant pumps. The

17.5 DESIGN FEATURES

17.5.3 REACTOR CORE

17.5.3.2 CONTROL ROD ASSEMBLIES AND DRIVE MECHANISMS (Cont'd)

primary motor-generator set is designed for a maximum rod release time of 7 seconds resulting from its coastdown, without any action by either Reactor Shutdown System.

- e. The maximum mechanical rod speed is approximately 70 in./minute above which the control rod drive mechanism roller nut will disengage by centrifugal force and release the rod.
- f. The maximum incremental secondary rod worth is 21¢/in.
- g. Series 1.5 and Series 2 Absorber Assemblies have the following lifetimes:
 - Row 3 Primary Rods-- 15.0×10^{22} n/cm² (E > 0.1 MeV) axial average fast neutron fluence delivered to the peak duct corner.* This limit applies for Row 3 primary rods operated in the shimmed configuration (≥ 30 in. withdrawn) and conservatively applies to full-out conditions (36 in. withdrawn) as well.
 - Row 5 Secondary Rods-- 15.0×10^{22} n/cm² (E > 0.1 MeV) axial average fast neutron fluence delivered to the peak duct corner.*

*Requires rotation per following prescribed schedule. Without rotation, the fluence limit is 11.8×10^{22} n/cm².

If First Rotation Performed at:	Then Second Rotation to be Performed at:
$5.0 \leq \phi t < 7.0$ - - - - -	$8.8 \leq \phi t \leq 10.8$
$7.0 \leq \phi t < 9.3$ - - - - -	$10.8 \leq \phi t \leq 12.8$
$9.3 \leq \phi t \leq 11.8$ - - - - -	Not Required

17.5 DESIGN FEATURES

17.5.3 REACTOR CORE

17.5.3.2 CONTROL ROD ASSEMBLIES AND DRIVE MECHANISMS (Cont'd)

- h. Series 1.5 and Series 2 absorber pins have the following design parameters:

	<u>Series 1.5</u>	<u>Series 2</u>
Cladding outside diameter	0.474 in.	0.474 in.
Cladding thickness	0.051 in.	0.046 in.
Pellet diameter	0.352 in.	0.362 in.
BOL diametral gap	0.020 in.	0.020 in.
Pellet density	92% TD	92% TD
Fill gas composition and pressure	helium, 1 atm at room temperature	

- i. Only one primary or secondary rod can be selected for manual withdrawal at any one time.

17.5 DESIGN FEATURES

17.5.4 REACTOR HEAT TRANSPORT SYSTEM

17.5.4.1 DESIGN PRESSURE AND TEMPERATURE

The Reactor Heat Transport System is designed for and shall be maintained in accordance with:

- a. The code requirements specified in FSAR Section 5B.1.5.
- b. The pressures and temperatures defined in Table 17.3.2-1.

17.5.4.2 DECAY HEAT REMOVAL

The three-loop configuration of the Reactor Heat Transport System is designed to assure continuous decay heat removal capability during all possible conditions including loss of power to the coolant pumps. Principal design features of the HTS which are relied upon for decay heat removal are summarized below:

1. Pony Motors and Power Supplies

Each primary and secondary coolant pump is provided with a pony motor. Pony motor power is supplied by redundant onsite power supplies and redundant offsite power supplies. Loop 1 is on the B10 swing bus (Emergency Power Subsystem 3), which is powered from either Emergency Power Subsystem 1 (which supplies Loop 2 pony motors on bus B4) or Emergency Power Subsystem 2 (which supplies Loop 3 pony motors on bus B5). See FSAR Section 8.2.2.1.3, Plant Power System.

2. DHX Controls

The heat removal rate of the DHX modules is controllable to match decay heat generation rate.

17.5 DESIGN FEATURES

17.5.4 REACTOR HEAT TRANSPORT SYSTEM

17.5.4.2 DECAY HEAT REMOVAL (Cont'd)

- 2.a Normal and emergency remote electrically operated controls are provided in the control room (See Section 17.5.6). Both electrical and pneumatic power are provided for damper operation.
- 2.b Local manual control of dampers, isolation gates, and secondary loop isolation valves is provided as a backup to remote DHX control. Local self-powered readout is provided (see Specification 17.3/4.2.7).
- 2.c Operator access for local DHX control has been demonstrated for all required situations including the DBE and tornado. Tornado hardening is provided for the East Loop DHX equipment. The DHX coarse dampers close fully on loss of electrical power, inlet vanes close to a preset position, and fine dampers are either closed to a preset position (if in the manual mode) or modulated (if in the automatic mode).

3. System Geometric Configuration

The spatial configuration of the HTS is a design feature that provides thermal driving head capability for primary and secondary natural circulation flow. Specifically:

- 3.a The geometric center of the effective tube length in the IHX is at least 15 ft above the reactor core geometric center, and
- 3.b The geometric center of the DHX tube bundle is a nominal 30 ft above the IHX geometric center. The as-built dimension is 29 ft, 7 3/4 in.

17.6 ADMINISTRATIVE CONTROLS

17.6.1 RESPONSIBILITY

Westinghouse Hanford Company, in its capacity as an owner's agent and operator of facilities at the Hanford Site, is responsible to the U.S. Department of Energy (DOE) for the safe operation of the Fast Flux Test Facility (FFTF).

17.6.2 PURPOSE

The purpose of the Technical Specifications is to define the limits/operating envelope within which the FFTF must be maintained. These limits are selected to maintain the integrity of the three barriers to release of radioactive materials (i.e., fuel cladding, HTS boundary, and containment building).

17.6.3 IMPLEMENTATION

Westinghouse Hanford is responsible for ensuring that the limits of the Technical Specifications are not exceeded. Meeting this responsibility includes implementation of all of the actions, surveillance, instrument operability, etc., necessary to ensure that these limits are not exceeded.

17.6.4 TECHNICAL SPECIFICATION VIOLATIONS AND NONCONFORMANCES

17.6.4.1 TECHNICAL SPECIFICATION VIOLATIONS

DEFINITION

A VIOLATION of the requirements of these specifications will be deemed to have occurred when:

1. The requirements of a "Safety Limit," Section 7.2.1, are not met,
2. The "Safety System Actuation Limits and TIME CONSTANTS," Section 17.2.2, are not met and the action is not invoked, or
3. An LCO of an "a" type (see Section 17.3.0.5.a) is not met.

17.6 ADMINISTRATIVE CONTROLS

17.6.4.1 TECHNICAL SPECIFICATION VIOLATIONS (Cont'd)

ACTION

In the event that a VIOLATION of the requirements of these specifications occurs, proceed as follows.

- a. Return the Plant to compliance as soon as possible and perform a SHUTDOWN of the affected operation, for example:
 - VIOLATION of LCO 17.3.2.1.a requires a REACTOR SHUTDOWN
 - VIOLATION of LCO 17.3.7.2.5 requires a termination of all fuel handling.
 - VIOLATION of Safety Limit 17.2.1.1, .2 or .3 requires a REACTOR SHUTDOWN and termination of all plant activities (including IEM Cell)
 - If an Actuation Limit is exceeded, only the affected operations require termination.
- b. Declare a Technical Specification VIOLATION.
- c. Immediately notify the DOE-RL and the DOE-HQ via the Occurrence Notification Center (ONC) of the VIOLATION.
- d. A written report, in the form of a Notification Report, shall be made within 24 hours.
- e. A written report on the matter, in the form of an Occurrence Report, shall be made in initial form within 10 working days.

17.6 ADMINISTRATIVE CONTROLS

17.6.4.1 TECHNICAL SPECIFICATION VIOLATIONS (Cont'd)

- f. When cause and effect of the VIOLATION have been determined and corrective action taken, the President, Westinghouse Hanford, will recertify to DOE-RL the readiness to resume operation. This certification shall document the bases and justification for resuming operation. Release to resume operation will be at the direction of DOE-RL.

17.6.4.2 TECHNICAL SPECIFICATION NONCONFORMANCES

DEFINITION

A NONCONFORMANCE of the requirements of these specifications will be declared when one or more of the following occurs.

1. The Design Feature Requirements (see Section 17.5) are not met.
2. The Administrative Control Requirements (see Section 17.6) are not met.
3. The ACTION statement is not invoked when it is discovered that the plant is outside the conditions specified by a "b" type LCO (see Section 17.3.0.5.b).
4. The Surveillance Requirements are not met (except because of instrument INOPERABILITY when the Action has been invoked per Section 17.4.0.1).

ACTION

In the event that a NONCONFORMANCE of the requirements of these specifications occurs, proceed as follows.

- a. The Action statement shall be invoked (unless compliance with the LCO has already been restored).

17.6 ADMINISTRATIVE CONTROLS

17.6.4.2 TECHNICAL SPECIFICATION NONCONFORMANCES (Cont'd)

- b. If the NONCONFORMANCE was a Surveillance Requirement not met, immediately perform the Surveillance or implement the Action.
- c. Immediately notify the DOE-RL and the DOE-HQ via the ONC of the NONCONFORMANCE.
- d. A written report, in the form of a Notification Report, shall be made within 24 hours.
- e. A written report in the form of an Occurrence Report shall be made within 10 working days describing the NONCONFORMANCE and corrective actions taken.

17.6.5 CHANGES TO THE TECHNICAL SPECIFICATIONS

All changes to the Technical Specifications shall be approved by DOE before implementation.

17.6.6 WAIVER TO THE TECHNICAL SPECIFICATIONS

Waivers to suspend various portions of the Technical Specifications when necessary for performance of special activities, such as acceptance testing or irradiation testing, may be granted. Waivers shall be approved by DOE. A waiver shall be for an explicit activity or defined time duration.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

BASES

17.2.1.4 FUEL STORAGE FACILITY (FSF)

Maintenance of the FSF storage vessel within the design envelope provides a barrier to release of fission products. The vessel was designed and constructed to the ASME B&PV Code, Section III, Class II. Design temperatures, pressures, and thermal transients were developed and used as the design events in the system analysis. The vessel design criteria do not require full operational capability following a faulted event but do require that all safety criteria be met (i.e., maintain fuel cladding temperature less than 1600 °F). The Safety Limit is appropriately set at the emergency level of accident severity. Details of the design criteria are given in FSF FSAR Chapters 3.1, 4.2, 4.3, and 15.6 (Ref. 72).

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.1 REACTOR SHUTDOWN SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

BASES

BACKGROUND/DERIVATION OF LIMITS

The primary function of the RSS is to protect the public from exposure to radioactive releases in excess of guidelines by scrambling the reactor, when necessary, to ensure the integrity of the barriers designed to prevent significant radioactive releases. The various actuation limits for the RSS (Table 17.2.2-1) have been selected to prevent certain plant process variables from reaching values that would endanger the integrity of the barriers (fuel cladding or primary coolant boundary) and to prevent exposure of plant personnel to excessive radiation.

The RSS is designed to normally operate in a two-out-of-three trip logic configuration. This configuration, with performance of required surveillance requirements, maintains the RSS reliability within the FSAR design goal. If a failed channel were left in service (not tripped), following its discovery, the reliability of the shutdown system to trip the plant could be reduced. Operation with a channel failed, but undetected, is within the bounds of the reliability analysis for the RSS.⁽¹⁶⁾

A one-out-of-two trip logic configuration (i.e., operating with one channel of a function tripped) is allowed for performing maintenance and surveillance. This configuration has a higher reliability for ensured reactor trip than the normal operating configuration but also has a lower operational reliability.

For operation at CDE operating conditions, the actuation limits in Table 17.2.2-1 are unchanged from normal FFTF operation except that the values of FULL FLOW and FULL POWER are redefined for CDE operation as specified in Sections 17.1.12 and 17.1.13. Since the FULL POWER CDE operating condition corresponds to an approximate 0.8 power-to-flow ratio (as compared to normal

TABLE B 17.2.2-1

REACTOR SHUTDOWN SYSTEM SETPOINTS

Functional Unit	System*	Setpoint
1. Manual Trip		OPERABLE
2. Reactor Overpower		
(A) Primary Wide-Range Nuclear - High	(P)	$\leq 112\%$ FULL POWER
(B) Secondary Flux/Increasing Delayed Flux	(S)	$\leq 13\%$ of FULL POWER (step increase) with ≥ 840 second time constant
(C) Primary Flux/Increasing Delayed Flux	(P)	$\leq 14\%$ of FULL POWER (step increase) with ≥ 840 second time constant
(D) High Startup Flux	(P) ^(b)	$\leq 1.5 \times 10^5$ cps
(E) Primary Wide-Range Nuclear - Low	(P) ^(a)	$\leq 48\%$ of FULL POWER
(F) Reactor Outlet Plenum Temperature	(S)	≤ 75 °F greater than steady-state operating temperature
3. Reactor Power/Flow Imbalance		
(A) Flux ² /Pressure	(P)	(see Equation 1)
(B) Flux/Reactor Flow	(S)	$\geq 83\%$ FULL FLOW at FULL POWER (see Equation 3)
(C) Pressure Permissive (permits bypass of high startup flux trip on power ascent above 60 kw)	(P)	(see Equation 2)
(D) Low Primary Loop Flow	(S) ^(c)	$\geq 60\%$ of FULL FLOW
(E) Loss of Offsite Power	(S) ^(c)	$\geq 75\%$ of Full Voltage (nominally 2.4 kV) ≤ 0.6 sec time delay
(F) Power Permissive	(S) ^(e)	$\leq 5\%$ FULL POWER

*See FOOTNOTES on page 17-266.

REACTOR SHUTDOWN SYSTEM SETPOINTS

Functional Unit	System*	Setpoint
4. Loss of Secondary Heat Removal Capability		
(A) IHX Primary Outlet	(P)	≤ 50 °F greater than steady-state operating temperature
(B) Low Secondary Loop Flow	(S) ^(d)	$\geq 55\%$ FULL FLOW.
5. High Primary Loop Flow	(S)	$\leq 112\%$ FULL FLOW
6. Reactor Vessel Coolant Level	(P)	≤ 4.4 inches below overflow level
7. Negative Rate of Power Change		
(A) Flux/Decreasing Delayed Flux	(P)	$\leq 11\%$ step decrease with ≥ 42.5 sec time constant
(B) Flux/Decreasing Delayed Flux	(S)	$\leq 11\%$ step decrease with ≥ 42.5 sec time constant
8. Experimenter Trip		
(A) PIOTA Temperature	(P) ^(f)	As specified for each test ^(f)
(B) Secondary Linear - Nuclear Power Level	(S) ^(f,g)	As specified for each test ^(f)
(C) Primary Power Level	(P) ^(a,f)	As specified for each test ^(f)

*FOOTNOTES:

- (P) is Primary Shutdown System; (S) is Secondary Shutdown System.
 (a) Bypass permitted when primary wide-range linear power channel indicates $>26\%$ FULL POWER.
 (b) Bypass permitted when primary range log power channel indicates $>1.5 \times 10^{-2}\%$ FULL POWER (60 kW) and pressure permissive (see Table B 17.2.2-1, Item 3C) is satisfied.
 (c) Bypass permitted when $<5\%$ power.
 (d) Bypass permitted when flow in corresponding primary loop is $<20\%$.
 (e) Automatically instates low primary loop flow and loss of offsite power trip functions at 5% FULL POWER.
 (f) This function may be bypassed or adjusted out of the operating range except when required for special reactor testing. When applicable, the SETPOINT and the plant conditions defining the applicability will be provided in a Technical Specifications waiver.

BASES 17.2.2.1

*FOOTNOTES: (Cont'd)

- (g) This function may be bypassed during power ascent when the reactor power exceeds the Low Power Permissive Setpoint, as specified when the function is active. Additionally, the function is automatically bypassed whenever all three primary HTS loops have flow in excess of 20% FULL POWER.

TABLE B 17.2.2.-, (Cont'd)
REACTOR SHUTDOWN SYSTEM SETPOINTS
SETPOINT EQUATIONS

EQUATION (1) - Flux ² /Pressure Trip $P \left[\frac{4}{G/P_F} \right] + 1.0 \leq K \left[\frac{(0.032 F)^2}{4} + X \right]$	EQUATION (3) - Flux/Reactor Flow % FULL FLOW - 0.8705 (% FULL POWER) $\geq$ -4.046
EQUATION (2) - Pressure Permissive (above 60 kW) $P \left[\frac{4}{G/P_F} \right] + 1.0 \leq K \left[1.538 + X \right]$	

X, K, P_F, G, P, and F are defined as follows:

$$X = 0.8067 \left(\frac{0.96 G - 25}{P_F} \right) - 0.060 \text{ volts}$$

$$K = 0.96 - \frac{\left(\frac{25}{G} \right)}{X + 0.060}$$

P_F = FULL FLOW pressure in psig*

G = full scale pressure instrument channel output in psig* (G = 1.25 P_F)

P = fraction of FULL FLOW pressure

F = percent FULL POWER

*Measured with respect to containment pressure at a specified tap.
Equation (2) is derived empirically to use gauge pressure.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.1 REACTOR SHUTDOWN SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

BASES

BACKGROUND/DERIVATION OF LIMITS (Cont'd)

FFTF FULL POWER operation) the FFTF driver fuel assemblies, other core components, and the primary coolant boundary will be exposed to less severe transient conditions.

Setpoints for each comparator of the RSS (Table B 17.2.2-1) determine the value, rate of change, or ratio between process variables at which a trip occurs. The setpoints have been specified so that an RSS trip will always occur before a parameter has reached its actuation limit even if instrument tolerances and channel drift combine in the most unfavorable way.

Reactor trips are provided to ensure that a scram mitigates the effects of events defined in the safety analyses of the FSAR, Section 15.1.1. In the case of anticipated and unlikely events, trips are provided in both the primary and secondary RSSs. Trips are provided to ensure that

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.1 REACTOR SHUTDOWN SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

BASES

BACKGROUND/DERIVATION OF LIMITS (Cont'd)

reactor Safety Limits are not exceeded. A list of initiating faults and functions that react to mitigate the results is given in FSAR Table 7.1-4 (Section 7.1).

The RSS CHANNEL TIME CONSTANT has been given a special definition in Section 17.1.5. This time interval plus the sensor time delay has been shown to be less than the function response time used in safety analysis for each function. The computer models used in the safety analysis were MELT-FCF213 and IANUS for reactivity transients and plant dynamics, respectively. Results recorded in Appendix F of the FSAR verify that the transients from each design basis event are terminated safely.

End-to-end simultaneous testing of the PPS systems is used where it is feasible. Where end-to-end testing is not feasible, testing in overlapping segments is acceptable. The RSS System is tested in segments with overlap so that all parts from the sensor to the control rods are checked. The sensor time response is determined by test where feasible, and, in installations where direct test is not possible, by a combination of data from suppliers and analysis.

Table 17.2.2-1 specifies the actuation limits and CHANNEL TIME CONSTANTS. These values are the "latest trip" values used in the FSAR Chapter 15 safety analyses. The latest trip values are those absolute values or times by which a trip must initiate. Additional allowances must be made in setting the setpoints and in compliance testing to account for the worst compounding of instrument and other errors. To perform valid response time testing, the trip setting must be at the nominal setting. The CHANNEL FUNCTIONAL TEST of the RSS electronics must use an acceptance criterion that makes allowance for error in untested channel segments and the error in measurement instruments.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.2 REACTOR HEAT TRANSPORT SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

BASES

BACKGROUND/DERIVATION OF LIMITS (Cont'd)

The associated pressure relief valve setpoints and their tolerances are given in Table B 17.2.2-2. It should be noted that in each case the relief valves are set to achieve their rated capacity at values less than the actuation limits shown in Table 17.2.2-2. The burst pressures of the rupture discs are also listed in Table B 17.2.2-2. The rupture discs ensure that the relief valves do not become blocked by sodium aerosols.

Also, the relief valve setting is below the rupture disc setting because the original secondary cover gas pressure was 75 psi vs the current 90 psi. Since the rupture disc is upstream of the relief valve, it is not necessary to revise the relief valve set pressure since the relief valve's primary function in this configuration is to remake the boundary after a pressure transient is relieved.

Note that a worst-case event can be postulated in which normal operating procedures and/or the Technical Specification limit on secondary pump discharge pressure (17.3.2.1) were ignored and then a cover gas control system failure occurred. The maximum possible secondary loop pressure that could be generated is:

Pump head at maximum speed (1177 rpm) with 3 modules on line:	173 psi
Cover gas pressure with relief valve at its actuation limit:	132 psi
Elevation difference, relief valve to pump discharge:	8 psi
Instrument inaccuracy	8 psi
Maximum loop pressure:	<hr/> 321 psi

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.2 REACTOR HEAT TRANSPORT SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

BASES

BACKGROUND/DERIVATION OF LIMITS (Cont'd)

The indicated pump discharge pressure for this event would be a 307 ± 8 psi and would, thus, require a REACTOR SHUTDOWN and SAFETY REVIEW before restart (see Sections 17.3.2.11 and 17.3.2.1). A one-time event of this magnitude would not cause failure of the secondary pressure boundary.

APPLICABILITY

As discussed in the Bases for Section 17.3.5.6, overpressure protection is required at all times.

ACTION

If pressure relief is not provided at or below the actuation limit, the affected piping system could be overstressed. If no relief path is available, the required actions reduce the system pressure and minimize the risk of a pressure excursion. This is discussed further in the Bases for Section 17.3.5.6.

SURVEILLANCE REQUIREMENTS

Required Surveillance for the relief valve actuation limits is discussed in Section 17.3.5.6.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.3 CONTAINMENT BUILDING - CONTAINMENT ISOLATION SYSTEM ACTUATION LIMITS

BASES

BACKGROUND/DERIVATION OF LIMITS (Cont'd)

The source term for the HCDA, which requires Principal Containment Isolation, is 30 to 100 times that of the head seal failure, per Appendix A of the FSAR. Therefore, the radiation field at the detector is also 30 to 100 times larger. Since this exceeds the upper bound of the radiation monitor range by at least an order of magnitude, the upper limit, 1 R/hr, was selected as the actuation limit.

CIS Time Response - The 3.0 second time limit for channel RESPONSE TIME is established, in conjunction with the 3.0 second limit on valve closure time, to match the minimum 6.0 second transit time from the H&V Containment Exhaust monitor location to the exhaust valve. Due to the inherent time delays in the diode pump circuit of the log ratemeter, however, the channel response time for a step change to a level only slightly above the actuation limit for H&V isolation will be longer than 3 seconds, approaching as long as one minute. If the source results in a step change to 20 mR/hr, or greater, the 3.0 second limit is met. The longer response time at the low levels is acceptable since the low level of activity results in no significant increase in the dose at the 1.5 mile control distance.

The CIS setpoints are set more conservative than the actuation limits to allow for instrument error and drift over the 30-day surveillance interval. The actual setpoints are shown in Table B 17.2.2-3.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.3 CONTAINMENT BUILDING - CONTAINMENT ISOLATION SYSTEM ACTUATION LIMITS

BASES

BACKGROUND/DERIVATION OF LIMITS (Cont'd)

- b. Vacuum Relief Actuation Limit - The building design pressure, with a safety factor of four, is -0.656 psig (-18.17 in. of water at 60 °F) to +10 psig. Containment relief valves provide under-pressure protection, and their actuation limits are based upon having the valves open before the building collapse pressure is reached.

Over-pressure protection is not provided because no design basis event was identified that caused building pressure to approach the design value. Likewise, over-temperature protection is not provided, because no design basis event was identified that would cause building temperature to exceed its design value of 250 °F. The containment vacuum relief actuation limit is given in Table B 17.2.2-4. Because these relief valves protect the containment structure, they need not be OPERABLE when an opening as large as the vacuum relief exists (e.g., both doors on any air lock are open) or equivalent relief protection is provided. The actual valve setpoint is more conservative to allow for valve setpoint inaccuracy and is shown in Table B 17.2.2-4.

- c. Pressure Actuated CIS Valve Actuation Points - Process lines containing air, nitrogen, and argon self-sensing pressure-actuated isolation valves isolate when the pressure in line falls below the setpoints. The 10-psig actuation limit is based upon isolating the lines if the system pressure falls below the design pressure of the containment building, thus ensuring no containment building atmosphere escapes through these lines. Actual minimum setpoints are 12 psig to allow for instrument inaccuracy.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.3 CONTAINMENT BUILDING - CONTAINMENT ISOLATION SYSTEM ACTUATION LIMITS

BASES

APPLICABILITY

The containment isolation function is applicable at any time that there is a potential for release of radioactivity in containment at a level that will require action from the CIS to prevent a release which could cause doses in excess of those in the matrix (paragraph 17.2.2.3). CONTAINMENT INTEGRITY is required as specified in Section a of the CONTAINMENT INTEGRITY Specification 17.3/4.4.1.

The In-Containment CAPS Blowers (ICCB) vent to the Containment H&V exhaust. The ICCB system may be used to maintain a vacuum on the CAPS vacuum header at times when CONTAINMENT INTEGRITY is not required. Therefore, the CIS H&V exhaust shall be OPERABLE as a backup to the ICCB RISH (see Specification 17.3.3.4.1) to shut off the ICCB exhaust in the event of a radiation release. If the H&V CIS does actuate due to a ICCB source, there are multiple shutdowns for the ICCB system (pressure sensor in H&V duct, RISH-52143, operating procedure, etc). Therefore, escape of significant amounts of radioactivity from containment penetrations is incredible and it is not appropriate to require CONTAINMENT INTEGRITY when the ICCB is OPERATING.

The containment is protected against external pressure by vacuum relief valves, which must be functional in all MODES since containment could be isolated at any time. An isolated containment is susceptible to an external pressure failure unless the vacuum relief system is OPERABLE.

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.3 CONTAINMENT BUILDING - CONTAINMENT ISOLATION SYSTEM ACTUATION LIMITS

BASES

ACTION

- a. All NON-OPERATING CIS channels shall be placed in the tripped condition to ensure the containment will isolate, given a failure in another channel. When one channel of a set of three redundant channels becomes nonfunctional, it is important that the system be placed in a configuration such that no single random failure can prevent completion of the protective function. This requirement is achieved by manually tripping the comparator for any channel that is not meeting the appropriate actuation limits and RESPONSE TIME specified in Table 17.2.2-3.

This action changes the scram logic from "2 out of 3" to "1 out of 2" because a trip of either one of the remaining channels will then cause a containment isolation.

- b. The containment building wall thickness has been sized on the basis of external loading and is vulnerable to collapse if the design vacuum is significantly exceeded. To protect the public, the plant needs to be configured to reduce to a minimum the need for CONTAINMENT ISOLATION.

Once CONTAINMENT INTEGRITY is not required, valves or air locks can be opened to create an alternate vacuum relief path to protect the containment from excessive external pressure.

- c. If a Containment Isolation Valve (CIV) is determined to not meet the actuation limit, CONTAINMENT INTEGRITY has been lost. CONTAINMENT INTEGRITY may be restored by closing and securing one of the CIVs in that

17.2 SAFETY LIMITS AND SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2 SAFETY SYSTEM ACTUATION LIMITS AND TIME CONSTANTS

17.2.2.3 CONTAINMENT BUILDING - CONTAINMENT ISOLATION SYSTEM ACTUATION LIMITS

BASES

ACTION (Cont'd)

line. The penetration must be isolated by a CIV because only the CIVs have been designed for and qualified to establish the containment boundary.

17.3/4.1 REACTOR

17.3/4.1.2 CORE LOADING CHARACTERISTICS

BASES

17.3.1.2 BACKGROUND/DERIVATION OF LIMITS

- a. The maximum lifetime of driver fuel pins is limited to ensure that the design limit of 0.2% total predicted inelastic strain or equivalent stress-based cumulative damage function (CDF) would not be exceeded during steady-state operation. This will then allow 0.1% clad strain for anticipated events and 0.4% for unanticipated events and still remain within the 0.7% allowance. The nominal coolant inlet temperature used in the calculation is 680 °F with a ± 15 °F variance caused by instrument and control uncertainties. Coolant inlet temperatures above this value would cause the fuel burnup limit to be reduced to maintain the 0.2% steady-state cladding strain limit. The calculated lifetime limits are related to the cladding strain limit (Ref. 1) through the method described in Reference 55. The lifetime is related to the axial-average total fission power and the assembly flow, since these parameters determine the fission gas production and the maximum cladding temperature. The equivalent BOL power used with Figure 17.3.1.2-1 must be selected as follows:
- The actual power history of the fuel must be bounded by a linearly decreasing power ramp starting at the equivalent BOL power and decreasing by 15.5% at 80,000 Mwd/MTM
 - For variations in assembly flow, both the absolute power and the power-to-flow ratio used to enter Figure 17.3.1.2-1 must bound the actual power and power-to-flow ratio of the assembly whose life-time is to be determined.

17.3/4.1 REACTOR

17.3/4.1.2 CORE LOADING CHARACTERISTICS

BASES

17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

- b.1 The phenomena that limit the absorber assembly lifetime are bottom plate-wear pad interaction, pellet-cladding interaction, internal gas pressure, relative bowing of the inner and outer duct components, and pin bundle-inner duct interaction. The following analyses⁽⁶⁾ are provided to support a Series 2 absorber lifetime surveillance fluence of 15.0×10^{22} n/cm². This is the axially averaged fast neutron fluence ($E > 0.1$ MeV) to the peak duct corner in the absorber assembly based on 2DB two-dimensional diffusion theory calculations performed in support of and reported in the Reload Design Report (RDR) for each cycle.

In some cases, data obtained from Series 1 absorbers are used to determine lifetime limits for the Series 2 absorbers. From the assembly vehicle viewpoint, the two designs are essentially identical. The major difference is in the absorber pin design. For the Series 2 design, the cladding was thinned slightly to provide a larger pellet-to-cladding gap and thereby extend the B₄C swelling limited lifetime. All the Series 1 absorbers have been either destructively examined or transferred to the FSF. There will be no further use of Series 1 absorbers in the FFTF.

A Series 1.5 absorber design also exists. The Series 1.5 absorbers result from the conversion of six fixed shim absorbers (Series 1 design) to movable absorber assemblies. This is accomplished by downloading the absorber pins and reloading the B₄C pellets into the appropriate cladding and assembly hardware for a moveable absorber assembly. During this conversion process, the B₄C pellets were ground down by 0.006 in. on the diameter to provide a 0.020-in-diametral pellet-cladding gap, the same as the Series 2 absorber design. The Series 1.5 and Series 2 absorber pin designs are functionally equivalent, except that the Series 1.5 pellet diameter is 0.010 in. smaller than the Series 2 pellet diameter, and the Series 1.5 absorber pin cladding is 0.005 in. thicker than the Series 2

17.3/4.1 REACTOR

17.3/4.1.2 CORE LOADING CHARACTERISTICS

BASES

17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

absorber pin cladding. This would have a minor impact on two of the lifetime limiting phenomena (i.e., pellet-cladding interaction and gas pressure loading of the cladding). In both cases, the impact is in the direction of being less limiting for the Series 1.5 design (i.e., the Series 1.5 lifetimes would be slightly extended). All other assembly design aspects of the Series 1.5 and Series 2 absorbers are identical. Therefore, the justifications presented below for a lifetime extension to 15.0×10^{22} n/cm² for the Series 2 absorbers also apply for the Series 1.5 absorbers.⁽⁹⁹⁾

Bottom Plate-Wear Pad Interaction--Differential swelling between the SA 316 SST bottom plate and the encircling Inconel* 718 wear band can potentially load the wear band at high fluence levels and possibly fracture it, which may inhibit scram performance. Postirradiation examination (PIE) was performed on the bottom plate and wear band components from the CRA-544 control rod irradiated to a surveillance fluence of 13.2×10^{22} n/cm² to determine actual swelling behavior and ductility characteristics. The CRA-544 assembly was initially a Series 1 absorber that was converted to a test after exceeding its Technical Specification limit to provide this needed data. The CRA-544 assembly was rotated once at the end of Cycle 9A.2 at a surveillance fluence of 9.4×10^{22} n/cm². Detailed analyses were subsequently performed⁽⁶⁾ to show that the stress/strain response of the wear band resulting from bottom plate swelling was within the demonstrated material capabilities. Based on the minimum ductility properties measured on the wear band specimens, it was determined that approximately 10% local strain can be accommodated without failure. At a surveillance fluence of

*Inconel is a trademark of Inco Alloys International.

17.3/4.1 REACTOR

17.3/4.1.2 CORE LOADING CHARACTERISTICS

BASES

17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

15×10^{22} n/cm², the calculated maximum local strain in the wear band resulting from differential swelling is approximately 5%, providing a safety margin of about 2.0 at this fluence. Thus, the lifetime limit for bottom plate-wear band interaction is $>15 \times 10^{22}$ n/cm².

Conservatism inherent in this evaluation is use of the minimum measured ductility in the CRA-544 wear pad, a conservative interpretation of the measured swelling data on the CRA-544 bottom plate,⁽⁶⁾ and a factor of 2 margin between predicted maximum strain and failure strain.

Pellet-Cladding Interaction--The boron carbide (B₄C) pellets swell faster than the 316 stainless steel cladding under irradiation, such that the pellet-cladding gap eventually closes. Further irradiation then results in pellet-cladding mechanical interaction (PCI), which can produce permanent strain in the cladding. If strain levels become substantial enough, cladding breach can result. Available data indicate that so long as PCI cladding strain levels remain below 2%, the cladding retains its integrity (i.e., does not breach). Table B 17.3.1.2-1 summarizes measured PCI results and consequences for stainless steel clad boron carbide pins.⁽⁶⁾

Table B 17.3.1.2-1. Pellet-Cladding Interaction Experience.

Experiment	Cladding	Clad T (°F)	Peak Strain	Breach
EBR-II/HEDL	304 SST	1000-1250	0.5%	No
EBR-II/HEDL	316 SST	1350	1.8%	No
EBR-II/HEDL	316 SST	1200	1.8%	No
EBR-II/HEDL	316 SST	1200	1.6%	No
EBR-II/HEDL	316 SST	800	1.3%	No
EBR-II/HEDL	316 SST	800	0.7%	No
FFTF/WHC	316 SST	1000	1.2%	No
BOR-60/USSR	OX16H15M3B	1000	0-2%	No
BOR-60/USSR	OX16H15M3B	1000	0-2%	Yes

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17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

The only reported occasion of cladding breach from PCI was by the Soviet Union. Control rod pins in the BOR-60 reactor were found to have developed longitudinal cracks in the cladding when strain levels were in the 2% to 4% range. This cladding was composed of an alloy similar to 316 SST in composition, except for molybdenum content. No significant loss of B_4C from the ruptured pins was noted. Absorber pins in the BOR-60 control rods with less than 2% cladding deformation did not breach.

It should be noted that extensive testing has shown that the consequences of an absorber pin breach are benign. The absorber pins do not contain fission gas, and the B_4C has been shown to not erode significantly with prolonged exposure to sodium. Nonetheless, the design goal for the Series 2 absorbers is to preclude cladding breach.

Based on the above information, a cladding strain level of 1% is defined as the lifetime limit for pellet-cladding interaction. This provides a factor of 2 margin to the minimum observed PCI strain level associated with cladding breach. The 1% strain value pertains to the cladding outside diameter (OD) measured at room temperature, the same as the conditions for the PCI measurement data presented above.

The PCI lifetime limit for Series 2 absorbers is determined by projection of PCI data obtained from a Series 1 absorber. Postirradiation examination of the CRA-548 absorber assembly irradiated to a surveillance fluence of 9.9×10^{22} n/cm² showed that the absorber pins had incurred a peak PCI of 0.17%. This was a Row 5, Series 1 absorber (0.014-in. pellet-to-cladding gap), but it nonetheless provides a tie point for

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17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

projecting PCI in Series 2 absorbers. (Several other Series 1 absorbers have been irradiated to higher exposures [e.g., up to 13.3×10^{22} n/cm² without problem] but lack of resources have precluded the examination of them.) With 0.17% PCI strain measured at 9.9×10^{22} n/cm², the remaining allowable strain to the 1% PCI limit is 0.83%. Since the pellet-cladding gap in the Series 2 absorbers is increased to 0.020 in., an additional 0.006 in. of diametral gap is also available to accommodate B₄C swelling before any PCI is incurred. In terms of PCI strain on the cladding OD, this additional gap translates into an added 1.00% allowable dilation. The allowable additional PCI strain on the cladding OD (relative to the CRA-548 tie point) then totals: $0.83 + 1.00 = 1.83\%$.

To project the end-of-life fluence from the CRA-548 tie point at 9.9×10^{22} n/cm², the PCI strain rate must be established. Data are available from postirradiation examination of Series 1 absorbers that can be used to determine the PCI strain rate. In addition to the measured PCI in CRA-548, B₄C pellet swelling data were obtained on two other Row 5, Series 1 absorber assemblies.⁽⁶⁾ It is also known that the onset of boron carbide swelling occurs at a surveillance fluence of about 1.0×10^{22} n/cm². These data are presented in Table B 17.3.1.2-2. The B₄C pellet swelling values are converted into strain values relative to the cladding inner diameter (ID). Because there was still a positive pellet-cladding gap in these pins, the converted strain values are negative. Based on conservation of mass (no volume change associated with creep or plastic strain), an adjustment factor must be applied to the cladding ID strain to convert to cladding OD strain. For the Series 1 absorber pins (0.474 in. OD $\times$ 0.372 in. ID), this factor is 1.618 (i.e., the cladding ID strain is divided by 1.618 to determine the

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17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

corresponding cladding OD strain). The cladding OD strain data in Table B 17.3.1.2-2 are plotted in Figure B 17.3.1.2-1 as a function of surveillance fluence. A linear regression analysis provided the fit shown in the figure. As indicated, the PCI strain rate based on these data is 0.28% per 10^{22} n/cm² surveillance fluence.

Table B 17.3.1.2-2. Data From Series 1 Absorbers
Used to Determine PCI Strain Rate.

Source	Fluence	Pellet ΔD	Clad ID Strain	Clad OD Strain
CRA-548	9.9	4.7%	--	0.17%*
CRA-537	8.2	3.4%	-0.49%	-0.30%
CRA-541	5.6	2.2%	-1.65%	-1.02%
B ₄ C ΔD Onset	1.1	0.0%	-3.76%	-2.33%

*Direct measurement

Pellet measurements performed on CRA-548 also substantiated the cladding strain measured by profilometry. The inferred cladding OD strain from pellet diameter measurements was +0.47%. This is slightly higher than the measured 0.17% cladding strain, but at the high swelling levels in these pellets (about 5% ΔD) this is not unexpected. Some of the diametral expansion in pellets at this swelling level is often compressible because of the development of cracks in the pellet structure. Therefore, the measured pellet swelling data from CRA-548, while not directly used in determining PCI strain rate, supports the validity of the cladding profilometry measurements.

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Figure B 17.3.1.2-1. PCI Strain Rate Projected by
Series 1 Measurement Data.

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The PCI strain is also calculated by the CNRD2 code,⁽⁶⁾ based on the B₄C swelling correlation incorporated in the code. Figure B 17.3.1.2-2 shows the predicted PCI strain behavior for the peak pin in both Series 1 and Series 2 absorbers in a Row 5 location. CNRD2 performs calculations in terms of equivalent full-power days (EFPD) at 400-MW operating conditions. Based on fluence calculations performed for the Reload Design Reports (RDR), 1 EFPD at these conditions is on the average equivalent to a surveillance fluence of 0.0225×10^{22} n/cm² in a Row 5 location and 0.027×10^{22} n/cm² in a Row 3 location. The strain rates shown in Figure B 17.3.1.2-2 are converted to a fluence basis using the Row 5 conversion factor. As indicated, the PCI rate for Series 2 is somewhat lower than for Series 1 (0.30% per 10^{22} n/cm² versus 0.35% per 10^{22} n/cm²). This is because at the relative fluence levels where PCI is first incurred, the B₄C swelling rate (on a fluence basis) for the Series 2 absorber is lower because of a reduced burnup rate in the B₄C from depletion of the boron-10 isotope.

Because the CNRD2 projected PCI strain rate is more conservative than that indicated by the Series 1 PIE data (0.30% versus 0.28% per 10^{22} n/cm²), it was selected to determine the lifetime fluence limit associated with the additional 1.83% allowable strain for the Series 2 absorber. The lifetime fluence limit for Row 5, Series 2 absorbers is thus:

$$\text{Fluence Limit} = 9.9 + 1.83/0.30 = 16.0 \times 10^{22} \text{ n/cm}^2$$

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Figure B 17.3.1.2-2. CNRD2 Projections of PCI Strain.

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BASES

17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

The PCI limits for Series 2 absorbers in Row 3 positions were obtained directly from CNRD2 analyses, which are conservative as indicated above. For a fully withdrawn Row 3 absorber, the surveillance fluence limit is 26.7×10^{22} n/cm². If the Row 3 rod were operated continuously in the shimmed configuration (≥ 30 in. withdrawn), the fluence limit is 20.0×10^{22} n/cm².

Conservatisms inherent in these evaluations include the 1% PCI strain limit (factor of 2 below minimum measured strains in breached pins) and the application of the CNRD2 PCI strain rate (higher than PIE measured indications). Considerable emphasis is placed on the CRA-548 measured PCI tie point (0.17% at 9.9×10^{22} n/cm²). The PCI measurement uncertainty is 0.12% at the two-sigma level. This would reduce the lifetime by 0.4×10^{22} n/cm² at the two-sigma uncertainty level, which is substantially less than the margin of 1.0×10^{22} n/cm² between the Row 5 nominal PCI lifetime and the Series 2 overall lifetime fluence limit of 15.0×10^{22} n/cm².

Internal Gas Pressure--Helium is produced in the neutron capture reaction in boron carbide, and some of this helium is released to the absorber pin plenum. The associated gas pressure buildup exerts a load on the cladding and can produce damage and strain in the cladding if high stress levels are reached.

Gas release rates from boron carbide are relatively low and are not limiting for the thick-walled Series 2 absorber pin design. The CNRD2 calculations were performed out to 1,000 EFPD at 400 MW conditions for both Row 3 and Row 5 locations. This corresponds to surveillance fluence

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17.3/4.1.2 CORE LOADING CHARACTERISTICS

BASES

17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

levels of 27×10^{22} n/cm² and 23×10^{22} n/cm² for Row 3 and Row 5, respectively. At these high exposure levels, the cumulative damage function (CDF) in the cladding from gas pressure loading was zero for both cases. Hoop stress in the Row 5 peak absorber pin at this exposure was 4,166 psi, while in the Row 3 absorber it was only 1,956 psi. These values compare to a yield strength of over 75,000 psi for the cladding at operating temperatures.

An additional conservatism inherent in this evaluation is that the gas release correlation employed in CNRD2 projects approximately twice the gas release quantities as those observed in postirradiation examinations of FFTF absorber pins.

Relative Duct Bowing--Thermal and flux gradients across the absorber assembly cause the inner duct and outer duct assemblies to bow as irradiation proceeds. Should the relative bowing between the two components become severe enough, interference to the vertical translation of the absorber bundle could be encountered. The end-of-life condition for this phenomenon is defined as the point where the relative bowing is such as to cause a forced contact situation between the movable inner duct assembly and the fixed outer duct. This condition corresponds to a configuration where the two wear pads contact the outer duct on one side of the assembly, while the inner duct tube contacts the outer duct on the opposite side of the assembly. This condition is illustrated in an exaggerated sense in Figure B 17.3.1.2-3. Such a situation could impede the vertical travel of the absorber bundle and potentially impact scram response. The full vertical translation (36-in. stroke) must be considered in evaluating this forced contact situation.

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Figure B 17.3.1.2-3. End-of-Life Condition from Bowing Interference.

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BASES

17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

Bowing interference is prevented by a rotation scheme where, after a period of bowing in one direction caused by flux and thermal gradients, the assembly is rotated 180° such that subsequent bowing in the opposite direction serves to straighten the assembly again. This rotation scheme has been successfully used on a number of control rods in FFTF, which operated well after rotation with no increase in scram time.

Bowing analysis of the Series 2 absorber assembly was performed using the CRPBOW code.⁽⁶⁾ The code performs calculations for steel swelling, irradiation-induced creep, thermal elongation, and beam centerline displacement as a function of operating time and pin bundle axial position. The flux and thermal gradients structured into the code represent FFTF Row 5 control rod conditions at 400 MW power operation. Because these gradients would be less at lower operating power levels (e.g., 291 MW), the evaluations are conservative for these cases (i.e., bowing predictions are exaggerated for equivalent fluence conditions). While the flux and temperature conditions intrinsic in the code are not applicable to Row 3 control rod analyses, the Row 5 lifetime capabilities in terms of fluence limits apply conservatively to Row 3 rods. This is because the flux and temperature gradients are less in a Row 3 location (e.g., the flux decreases about 20% across a Row 5 rod and about 13% across a Row 3 rod). Also, the absorber bundle in a Row 3 rod is normally withdrawn during reactor operation and, therefore, not as subject to bowing tendencies.

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17.3/4.1.2 CORE LOADING CHARACTERISTICS

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17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

To better reflect actual fluence accumulation during each cycle at 291-MW power operation, CRPBOW was run on the basis of 80 EFPD cycles at 400-MW power conditions. As indicated above, this conservatively applies more severe flux and temperature gradient conditions than actually exist at 291 MW operation, while accruing approximately the correct fluence during each operating cycle at 291 MW. It is assumed that the control rod is steadily withdrawn from 16 in. out to 36 in. out during the course of each cycle.

The CRPBOW performs calculations on the basis of EFPD at 400 MW operating conditions. The following factors are used to convert operating time to Row 5 surveillance fluence values: 0.0225×10^{22} n/cm² per EFPD on the side facing core center (Side 1) and 0.0180×10^{22} n/cm² per EFPD on the side facing away from core center (Side 2). These conversion factors were obtained from RDR calculations performed for the FFTF during the period it operated at 400 MW (Cycles 1 through 8).

Analyses were performed for a number of cases where the control rod was rotated at various times (at the end of incremental 80 EFPD cycles). Figure B 17.3.1.2-4 shows typical CRPBOW projections of bowing behavior. In this particular case, the control rod is rotated at 400 EFPD. Side 1 corresponds to the side facing toward core center, with Side 2 representing the opposite side of the absorber assembly. It is seen that once the control rod is rotated, the bowing trends reverse and the clearance on Side 1 begins increasing. Eventually, as bowing continues in this reverse direction, clearance on the opposite side (Side 2) begins decreasing and ultimately becomes limiting if no further rotation is performed. On the basis of the CRPBOW clearance parameter plotted in the

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Figure B 17.3.1.2-4. Bowing Behavior with Rotation at 400 EFPD.

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BASES

17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

figure, a value of -0.100 in. corresponds to the three-point forced contact situation that represents end of life. This condition is reached on Side 2 at an exposure of 710 EFPD. The cumulated surveillance fluence at this point is 14.6×10^{22} n/cm². This value accounts for the reduced fluence rate on Side 1 after the assembly is rotated. The total fluence on Side 1 remains greatest at this point. The corresponding Side 2 fluence level is 14.2×10^{22} n/cm².

If the assembly is never rotated, the surveillance fluence limit on Side 1 is 11.8×10^{22} n/cm². That is, a three-point forced contact situation is reached at this point resulting from Side 1 bowing distortions. With rotation, bowing interference on Side 2 eventually becomes limiting (if the assembly is rotated only once).

Figure B 17.3.1.2-5 presents this single rotation bowing lifetime limit as a function of the surveillance fluence at the time of rotation. The lifetime fluence limits shown account for the reduced fluence accumulation rate on the limiting side after rotation. A straight line fit to the data is included in the figure.

In evaluating the CRPBOW results, it is noted that when the assembly is in the bowing mode (surveillance fluence greater than about 6×10^{22} n/cm²), the opposite side bowing does not become limiting after rotation until the peak surveillance fluence (side towards core center) exceeds an additional increment of 7×10^{22} n/cm². This is true for any rotation fluence of 5×10^{22} n/cm² or greater. This is the basis for developing the fluence ranges for the second rotation presented in Table B 17.3.1.2-3. With assembly rotations performed within the specified fluence ranges, the bowing-limited lifetime is at least

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Figure B 17.3.1.2-5. Bowling Lifetime Limit with Single Rotation.

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BASES

17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

15×10^{22} n/cm². It is apparent that with appropriate rotation sequences, duct bowing need never become limiting. However, the rotation schedule specified in Table B 17.3.1.2-3 ensures a bowing-limited fluence capability of at least 15×10^{22} n/cm².

A number of conservatisms are included in the bowing analyses. Application of 400-MW conditions to 291-MW operation imposes exaggerated flux and temperature gradients across the assembly (i.e., bowing responses would actually be less at lower power conditions). The two-sigma version of the stainless steel swelling correlation (Nuclear Systems Materials Handbook, Rev. 9) was employed to conservatively project bowing behavior. Although Row 3 rod analyses were not specifically performed, the Row 5 results conservatively apply to Row 3 limits with respect to fluence capabilities because of the reduced gradients in Row 3 and the withdrawn mode of operation for the Row 3 absorber bundle.

Pin Bundle--Inner Duct Interaction--Caused primarily by creep strain incurred in the absorber pins, the pin bundle grows faster with irradiation than does the inner duct tube that contains the pin bundle. Thus, the clearance (0.026 in. at fabrication) between the pin bundle and inner duct diminishes with irradiation exposure. Once the pin bundle growth has closed this gap, further pin bundle dilation could cause the inner duct to expand and decrease the clearance between inner and outer ducts. This in turn could impact the bowing limited lifetime. While such pin bundle dilation could be easily accommodated by appropriate analytical bowing evaluations and control rod rotation schedules, a zero clearance between the pin bundle and inner duct tube is adopted as a conservative limit.

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17.3/4.1.2 CORE LOADING CHARACTERISTICS

BASES

17.3.1.2 BACKGROUND DERIVATION OF LIMITS (Cont'd)

CNRD2 calculates the bundle-to-duct clearance as function of operating exposure. For a Row 5 control rod, the lifetime fluence limit (zero remaining clearance) is 17.8×10^{22} n/cm². For a Row 3 rod, the fluence limit is 23.2×10^{22} n/cm².

The zero clearance criterion as a lifetime limit is very conservative, inasmuch as considerable inner duct dilation could be accommodated without impacting bowing limited lifetimes.

Table B 17.3.1.2-3. Series 2 Absorber Assembly Lifetime Limits.

Parameter	Fluence (10^{22} n/cm ²)	
	Row 3 Rods*	Row 5 Rods
Bottom Plate-Wear Band Interaction	>15.0	>15.0
Pellet-Cladding Interaction	20.0	16.0
Internal Gas Pressure	>27	>23
Relative Bowing	>15**	>15**
Pin Bundle-Inner Duct Interaction	23.2	17.8

*The values presented for the Row 3 rods reflect operation in the shimmed configuration (≥ 30 in. withdrawn), but conservatively apply to full-out conditions (36 in. withdrawn) as well.

**Requires rotation per following prescribed schedule. Without rotation, the fluence limit is 11.8×10^{22} n/cm².

If First Rotation Performed at:	Then Second Rotation to be Performed at:
$5.0 \leq \phi t < 7.0$ - - - - -	$8.8 \leq \phi t \leq 10.8$
$7.0 \leq \phi t < 9.3$ - - - - -	$10.8 \leq \phi t \leq 12.8$
$9.3 \leq \phi t \leq 11.8$ - - - - -	Not Required

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17.3/4.1.2 CORE LOADING CHARACTERISTICS

BASES

ACTION (Cont'd)

the FTRIA measurements obtained at FULL POWER (Reference 54). This provides the technical basis for using predicted values to ensure that the limits will be met on startup.

During reactor operation, the FTRIA and temperature data are evaluated twice during each normal CYCLE. If this evaluation indicates that a limit is being exceeded, a REACTOR SHUTDOWN will be initiated to a power level at which the limits are not exceeded. Because these limits do not represent an immediate safety hazard, a power reduction is permitted. A SAFETY REVIEW is then performed to assess the consequences of the ΔT condition and its impact on completion of the reactor CYCLE.

17.4.1.2 SURVEILLANCE REQUIREMENTS

a&b The irradiation history of core assemblies is determined and their allowable lifetimes are calculated as each CYCLE of operation is completed and before the start of the next CYCLE. In this way, core component accumulated exposure and fast fluence can be established at the beginning of each CYCLE and projected to the designed end of each CYCLE.

Fuel pin lifetimes given in Figure 17.3.1.2-1 are conservatively established to cover the worst-case conditions. An explicit calculation, using the same method that was used to prepare the figure, is permitted to allow individual conditions to be considered. Additionally, the past history of the fuel pin is to be accounted for to cover the "normal" operations (where the full hot channel factor allowances did not apply) or any unanticipated occurrence that may affect lifetime. (See Reference 55). A detailed description of the procedure used to implement the method developed in Reference 55 for explicitly evaluating driver fuel lifetimes is included in Reference 98.

17.3/4.1 REACTOR

17.3/4.1.2 CORE LOADING CHARACTERISTICS

BASES

17.4.1.2 SURVEILLANCE REQUIREMENTS (Cont'd)

An example of the use of this methodology (that provided the basis for permitting the DE4-1 assembly to successfully achieve a burnup beyond the fuel pin lifetime curve) is provided in Reference 56. Results from explicit calculations that were used to form the bases for determining individual fuel pin lifetimes are included as attachments to the Reload Design Report (RDR) for each CYCLE. They are overchecked by the Safety group in the Reload Verification Report (RVR).

The highest fluence experienced by a Series I DFA fuel pin, in the inner enrichment zone, at the end-of-life limit defined by Figure 17.3.1.2-1 is $\sim 17.5 \times 10^{22}$ n/cm² (E $\geq$ 0.1 MeV). This occurs for an average pin fission power of ~ 10.1 kW/ft. Series II DFAs and outer zone fuel have higher enrichments, thus lower flux for a given power; hence the Series I fuel is the bounding condition. Pin burnup lifetime is reached before the recent duct fluence database value of 17.5×10^{22} n/cm² (E $\geq$ 0.1 MeV) is reached. When fuel pin lifetime limits are obtained by performing an explicit calculation, a calculation of the associated duct fluence is also required. If the fluence exceeds this recent data base value (where the effects of changes in material properties on the duct functional performance are known), the material properties shall be reevaluated and shown to not be further degraded (i.e., the saturation values continue to apply). If degradation exists, the original duct design shall be reevaluated to determine whether the conclusions on duct performance remain valid for the revised duct material properties.

In support of the Reload Design Report (RDR) for each CYCLE, subassembly functional capabilities are verified in core component lifetime evaluations. Forces exerted on candidate ducts are evaluated to ensure that duct stresses are acceptable throughout the entire CYCLE.

17.3/4.5 PLANT SYSTEMS

17.3/4.5.6 PRESSURE RELIEF PATHS

BASES

17.3.5.6 BACKGROUND/DERIVATION OF LIMITS (Cont'd)

The vacuum relief systems for the containment building are designed and constructed in accordance with ASME, Section III, Class 2, Subarticle NA-3250. These valves are automatic relief valves, which are actuated by a differential pressure. These valves open at a predetermined setpoint and allow atmospheric air to enter the containment building to relieve an excessive negative pressure condition. This condition may result if the containment building is isolated after the containment atmosphere has first expanded because of heating and then contracts upon cooling, by malfunction of the containment pressure control system, or by accidental closure of an isolation valve in the ventilation inlet line during system operation. Detailed structural analysis,⁽³⁹⁾ based on a safety factor of four, indicates that even considering wind loads, the containment building will withstand a negative pressure of 0.656 psig (18.17 in. of water at 60 °F). The maximum negative pressure capability of the ventilation system is 16 in. of water. Structural integrity of the RCB is, therefore, not in question.

A pressure relief path may not properly perform its intended function should any of the following conditions occur:

17.3/4.5 PLANT SYSTEMS

17.3/4.5.6 PRESSURE RELIEF PATHS

BASES

17.3.5.6 BACKGROUND/DERIVATION OF LIMITS (Cont'd)

- The pressure relief valve does not meet its actuation limit as defined in Table B 17.2.2-2. This may have been demonstrated either during the periodic set pressure test or during an HTS pressure excursion. Failure to meet this limit implies that the relief system will not restrict the rise in system pressure during a transient to less than ten percent above the design pressure at the corresponding design temperature. This would be in conflict with the ASME code criterion and may compromise the integrity of the HTS pressure boundary.
- The vacuum relief valve does not meet its actuation limits as designed in Table B 17.2.2-4.
- The rupture disk associated with a functional pressure relief valve is either leaking or not intact. As previously discussed, this could foul the relief valve and preclude its operation.
- Inadvertent blockage or restriction of the pressure relief path would preclude proper functioning of the relief system. The associated upstream block valve must therefore be fully open to preclude inadvertent isolation and/or adding flow impedance which may limit the system relief capacity below the rated value. In addition, plugging either from frozen sodium condensing from the sodium vapor in the cover gas or from the formation of sodium hydride at local cold spots in the piping must be prevented.

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20.6 TEMPORARY DEVIATIONS

Temporary Deviations to the Criticality Prevention Specifications shall be placed in this section during the time period for which they are effective.

Number	ECN	Date Implemented	Date Removed
TD W405-1	A-19299	11/5/82	2/1/83
TD W405-1-1	A-19674	4/4/83	5/13/83
TD W405-2-1	441067	3/4/88	5/18/88
TD W405-2-2	113924	3/24/89	4/7/89
TD 405-1-2	108765	6/1/89	9/26/89
TD 405-2-3	108765	6/1/89	9/26/89
TD 405-2-4	175037	10/16/92	12/15/92

checked to locate and verify the abnormal flow meter reading.* The thermocouple readings for that assembly will next be checked. If two or three of these are reading higher than normal (but not necessarily greater than the alarm points), reactor power will be reduced until the reading for that point is in the operational range. Using the operational history of the affected assembly, the situation will be assessed to determine the probability of fuel pin failures and/or flow blockage. Based on this assessment, a decision will be made as to whether the reactor will be shut down for assembly removal or operation will continue at a reduced power level.

If the thermocouple readings are normal for at least two of the sensors, then the most probable cause of the alarm is an instrument malfunction. In order to verify this, the operational history of the flowmeter is reviewed and the electronics of the instrument are calibrated. If the flowmeter reading remains lower than the alarm point after calibration, the computer processing of its input will be suspended and operation continued.

Low flow in individual subassemblies before startup can be detected by manually scanning the driver fuel assembly flowmeter outputs. At this time there is no appreciable core temperature rise to observe with the thermocouples. By carefully observing and comparing flowmeter readings from one startup to another, any unexplained changes in assembly flow rate (which may be postulated to result from a large blockage) will be detected.

Because of the modular construction on the instrumentation, if the cause of the malfunction is not directly related to the sensor (e.g., a power supply failure), the repair can be made during reactor operation. However, maintenance of the flowmeters and thermocouples will be confined to outages in all instances.

C.11.2 FUEL FAILURE MONITORING

The Fuel Failure Monitoring (FFM) System provides the equipment necessary to detect, characterize, locate, and verify fuel and absorber pin failures. This

* Since Appendix C is concerned only with local faults, it will be assumed that only one flowmeter is reading out-of-range. However, procedure SI-31-7 (see SDD-31, Section 4.3.2.3.2) is also applicable when more than one flowmeter is reading low.

equipment is illustrated in Figure C.11-4 and includes sensors, associated signal conditioning equipment, read-out equipment, and annunciator units necessary to accomplish the following:

1. Monitor the reactor cover gas (SDD-31 and 82) for gaseous fission product activity and indicate abnormal activity levels (Reactor Cover Gas Monitoring System)
2. Monitor the reactor cover gas for activated tag gas released from failed fuel and absorber assemblies and indicate high activity levels (Reactor Cover Gas Monitoring System)
3. Monitor the reactor primary sodium coolant (SDD-51) for delayed neutrons and indicate abnormal activity levels (Reactor Delayed Neutron Monitoring System)
4. Monitor each closed loop system (SDD-61) for fission gas and delayed neutrons and indicate high activity levels. (Closed Loop Cover Gas Monitoring System and Closed Loop Delayed Neutron Monitoring System)
5. Locate failed fuel and absorber assemblies (Failure Location System)
6. Provide for sampling and analysis of reactor and closed loop primary sodium and cover gas (SDD-85).

The Fuel Failure Monitoring System is not a single integrated system. The system comprises several independent subsystems and analytical procedures providing data on failure occurrence, failure characteristics, and location of the failure to plant operations personnel. Since the reactor plant is designed to operate safely with as much as 1% of the core fuel leaking fission gases, fuel failure signals from FFM equipment are not used or needed by the Plant Protection System (SDD-99) to initiate reactor scrams.

Operation of the monitoring equipment is automatic and data is recorded through the use of magnetic tape and the DDH & DS (SDD-91). Annunciators are provided in the Reactor Control Room to alert the operator if any channel exceeds a preset level.

C.11.2.1 Reactor Cover Gas Monitoring System

The Reactor Cover Gas Monitoring System monitors the reactor cover gas for gaseous fission products that are normally present as tramp background and for gaseous fission and activation products that escape from failed fuel and absorber pins, respectively. The purposes of this system are: (1) to detect fuel and absorber pin failures and (2) to help characterize the failure as to type and degree.

Following a breach in cladding integrity, fuel and absorber pins release their contained gases to the sodium. Gases escape from the sodium coolant into the reactor cover gas. Background quantities are formed from neutron activation of the sodium and from fissioning of impurities. These impurities are uranium, plutonium, and thorium in the sodium and reactor core components and fuel contamination on the cladding. The monitoring system is designed to distinguish between the background radioactive gases and the gases released upon fuel pin failure. A reliable indication of a failure is essential to be able to make use of the Gas Tag Sampling Trap Failure Location System during the limited time (2 to 6 hours) before the escaped tag gases have been removed from the cover gas by the Radioactive Argon Processing System (RAPS).

The Reactor Cover Gas Monitoring System is composed of three major subsystems:

1. Cover Gas Sampling Subsystem (CGSS).
- 2&3. Two identical-in-function Fission Gas Monitoring Subsystems.

The Fission Gas Monitoring Subsystems scan the sample column with a pure germanium (Ge) diode gamma detector. Three specific fission gas isotopes (^{133}Xe , ^{135}Xe , and ^{85m}Kr) and one activation gas isotope (^{125}Xe) will be individually monitored. Absorber pin failures will be detected by measuring ^{125}Xe , formed by the activation of ^{124}Xe , which is a constituent of the tag gas used for absorber pin failure locations. An alarm in the Reactor Control Room will be activated in the event of an abnormal activity level for any of these isotopes.

NOTE: The Reactor Fission Gas Monitoring System cannot detect fuel failures after the reactor has been shut down for greater than 65 days, because of decay of the radioactive isotopes in the fuel to below the system's detection limits.

The Fission Gas Monitoring Subsystems also analyze data collected from the Ge diode gamma detector and the FGMS module control panels during a fuel or absorber pin failure to determine the basic characteristics of the failure (e.g., magnitude and burnup). More information on these subsystems is presented in Section C.11.2.3.1.

The time interval between the instant a cladding failure occurs (a burst-type gas release) and the time the operator is made aware of the situation will be less than 14 minutes. The majority of this increment is attributed to the gas transport time from the sodium surface to the reactor head outlet nozzle (2 minutes maximum) and from the nozzle to the location of the Ge diode detectors (10 minutes* maximum). The minimum detectable quantities of the various isotopes to be monitored are listed in SDD-94.

C.11.2.2 Reactor Delayed Neutron Monitoring System

The Reactor Delayed Neutron Monitoring System consists of an assembly of neutron detectors mounted in a shielding and moderator assembly situated adjacent to each of the three Reactor Heat Transport System (SDD-51) hot leg pipes. Identical instrumentation channels containing signal conditioning, display, and alarm equipment are provided for each of the three HTS loops. Coolant sodium transported past the detector assemblies is monitored for delayed neutrons emitted by decay of radioactive precursors in the sodium. The maximum transport time of the sodium from the core to the pipe section where the delayed neutron monitors are located is ~55 seconds at full flow.

* This interval is provided so that the ^{23}Ne can decay. This allows detection of the xenon and cesium isotopes whose peaks would otherwise be undetectable as they fall within the energy band of the Compton scattered electrons formed by the high-energy gamma of ^{23}Ne .

Generation of the delayed neutron precursors essentially ceases with reactor shutdown. Because of the lack of delayed neutron precursors during reactor shutdown, the Delayed Neutron Detection System will not detect cladding failures.

The detection system for each leg of the Heat Transport System (HTS) consists of an electrostatically shielded assembly of three BF_3 -filled gas proportional neutron detectors mounted in the shielding and moderator assembly

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OPERATION AT 680 °F INLET TEMPERATURE
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As a further conservatism, this analysis was based on the earlier Rev. 4 swelling equation. The longer incubation period associated with the Rev. 5 swelling equation results in less negative clearance at the maximum burnup.

F.3.3.3 Fuel Pin Life

Since swelling does not enter into the fuel pin design life calculations, temperature effects on the strength properties of the Type 316, 20% CW, stainless steel cladding are of primary interest. Design life of the fuel pin was conservatively set at 80 MWd/kg for a reactor inlet temperature of 600 °F. This margin is depleted at about 700 °F inlet temperature, and operation of the reactor with an inlet temperature in excess of 700 °F may lower the allowable peak burnup to less than 80 MWd/kg. A design margin of approximately 5 MWd/kg is available at 680 °F. This conclusion is supported by a large database illustrated in Figure F.3-8, which represents the population of EBR-II irradiation test burnup achieved for a large variety of peak cladding temperatures. Cladding temperatures are related to inlet temperatures to derive the foregoing conclusions. Operation past 80 MWd/kg is also acceptable for a range of power levels (see Addendum 1 to Appendix F).

F.3.4 ABSORBER ASSEMBLY DISTORTION

The phenomena that limit absorber assembly lifetime are bottom plate-wear pad interaction, pellet-cladding interaction, internal gas pressure, relative bowing of the inner and outer duct components, and pin bundle-inner duct interaction. The following analyses⁽⁶⁾ are provided to support a Series 2 absorber lifetime surveillance fluence of 15.0×10^{22} n/cm². This is the axially-averaged fast neutron fluence ($E > 0.1$ MeV) to the peak duct corner in the absorber assembly based on 2DB two-dimensional diffusion theory calculations performed in support of and reported in the Reload Design Report (RDR) for each cycle.

In some cases, data obtained from Series 1 absorbers are used to determine lifetime limits for the Series 2 absorbers. From the assembly vehicle viewpoint, the two designs are essentially identical. The major difference is

in the absorber pin design. For the Series 2 design, the cladding was thinned slightly to provide a larger pellet-to-cladding gap and thereby extend the B_4C swelling limited lifetime. All the Series 1 absorbers have been either destructively examined or transferred to the FSF. There will be no further use of Series 1 absorbers in the FFTF.

A Series 1.5 absorber design also exists. The Series 1.5 absorbers result from the conversion of six fixed shim absorbers (Series 1 design) to movable absorber assemblies. This is accomplished by downloading the absorber pins and reloading the B_4C pellets into the appropriate cladding and assembly hardware for a moveable absorber assembly. During this conversion process, the B_4C pellets were ground down by 0.006 in. on the diameter to provide a 0.020-in.-diametral pellet-cladding gap, the same as the Series 2 absorber design. The Series 1.5 and Series 2 absorber pin designs are functionally equivalent, except that the Series 1.5 pellet diameter is 0.010 in. smaller than the Series 2 pellet diameter, and the Series 1.5 absorber pin cladding is 0.005 in. thicker than the Series 2 absorber pin cladding. This would have a minor impact on two of the lifetime limiting phenomena (i.e., pellet-cladding interaction and gas pressure loading of the cladding). In both cases, the impact is in the direction of being less limiting for the Series 1.5 design (i.e., the Series 1.5 lifetimes would be slightly extended). All other assembly design aspects of the Series 1.5 and Series 2 absorbers are identical. Therefore, the justifications presented below for a lifetime extension to 15.0×10^{22} n/cm² for the Series 2 absorbers also apply for the Series 1.5 absorbers.⁽⁹⁾

Bottom Plate-Wear Pad Interaction--Differential swelling between the SA 316 SST bottom plate and the encircling Inconel 718 wear band can potentially load the wear band at high fluence levels and possibly fracture it, which may inhibit scram performance. Postirradiation examination (PIE) was performed on the bottom plate and wear band components from the CRA-544 control rod irradiated to a surveillance fluence of 13.2×10^{22} n/cm² to determine actual swelling behavior and ductility characteristics. Detailed analyses were subsequently performed⁽⁶⁾ to show that the stress/strain response of the wear band resulting from bottom plate swelling was within the demonstrated material capabilities. Bases on the minimum ductility properties measured on the wear band specimens, it was determined that approximately 10% local strain can be

accommodated without failure. At a surveillance fluence of $15 \times 10^{22} \text{ n/cm}^2$, the calculated maximum local strain in the wear band resulting from differential swelling is approximately 5%, providing a safety margin of about 2.0 at this fluence. Thus, the lifetime limit for bottom plate-wear band interaction is $>15 \times 10^{22} \text{ n/cm}^2$.

Conservatism inherent in this evaluation is use of the minimum measured ductility in the CRA-544 wear pad, a conservative interpretation of the measured swelling data on the CRA-544 bottom plate, and a factor of 2 margin between predicted maximum strain and failure strain.

Pellet-Cladding Interaction--The boron carbide (B_4C) pellets swell faster than the 316 stainless steel cladding under irradiation, such that the pellet-cladding gap eventually closes. Further irradiation then results in pellet-cladding mechanical interaction (PCI), which can produce permanent strain in the cladding. If strain levels become substantial enough, cladding breach can result. Available data indicate that so long as PCI cladding strain levels remain below 2%, the cladding retains its integrity (i.e., does not breach). Table F.3-4 summarizes measured PCI results and consequences for stainless steel clad B_4C pins.⁽⁶⁾

The only reported occasion of cladding breach from PCI was by the Soviet Union. Control rod pins in the BOR-60 reactor were found to have developed longitudinal cracks in the cladding when strain levels were in the 2% to 4% range. This cladding was composed of an alloy similar to 316 SST in composition, except for molybdenum content. No significant loss of B_4C from the ruptured pins was noted. Absorber pins in the BOR-60 control rods with less than 2% cladding deformation did not breach.

Based on this information, a cladding strain level of 1% is defined as the lifetime limit for pellet cladding interaction. This provides a factor of 2 margin to the minimum observed PCI strain level associated with cladding breach. The 1% strain value pertains to the cladding outside diameter (OD) measured at room temperature, the same as the conditions for the PCI measurement data presented above.

The PCI lifetime limit for Series 2 absorbers is determined by projecting PCI data obtained from a Series 1 absorber. Postirradiation examination of the CRA-548 absorber assembly irradiated to a surveillance fluence of 9.9×10^{22} n/cm² showed that the absorber pins had incurred a peak PCI of 0.17%. This was a Row 5, Series 1 absorber (0.014 in. pellet-to-cladding gap), but it, nonetheless, provides a tie point for projecting PCI in Series 2 absorber. With 0.17% PCI strain measured at 9.9×10^{22} n/cm², the remaining allowable strain to the 1% PCI limit is 0.83%. Since the pellet-cladding gap in the Series 2 absorbers is increased to 0.020 in., an additional 0.006 in. of diametral gap is also available to accommodate B₄C swelling before any PCI is incurred. In terms of PCI strain on the cladding OD, this additional gap translates into an added 1.00% allowable dilation. The allowable additional PCI strain on the cladding OD (relative to the CRA-548 tie point) then totals: $0.83 + 1.00 = 1.83\%$.

To project the end-of-life fluence from the CRA-548 tie point at 9.9×10^{22} n/cm², the PCI strain rate must be established. Data are available from postirradiation examination of Series 1 absorbers that can be used to determine the PCI strain rate. In addition to the measured PCI in CRA-548, B₄C pellet swelling data were obtained on two other Row 5, Series 1 absorber assemblies.⁽⁶⁾ It is also known that the onset of B₄C swelling occurs at a surveillance fluence of about 1.0×10^{22} n/cm². These data are presented in Table F.3-5. The B₄C pellet swelling values are converted into strain values relative to the cladding inner diameter (ID). Since there was still a positive pellet-cladding gap in these pins, the converted strain values are negative. Based on conservation of mass (no volume change associated with creep or plastic strain), an adjustment factor must be applied to the cladding ID strain to convert to cladding OD strain. For the Series 1 absorber pins (0.474 in. OD x 0.372 in. ID), this factor is 1.618 (i.e., the cladding ID strain is divided by 1.618 to determine the corresponding cladding OD strain). The cladding OD strain data in Table F.3-5 are plotted in Figure F.3-9 as a function of surveillance fluence. A linear regression analysis provided the fit shown in the figure. As indicated, the PCI strain rate based on these data is 0.28% per 10^{22} n/cm² surveillance fluence.

Pellet measurements performed on CRA-548 also substantiated the cladding strain measured by profilometry. The inferred cladding OD strain from pellet diameter measurements was +0.47%. This is slightly higher than the measured 0.17% cladding strain, but at the high swelling levels in these pellets (about 5% ΔD) this is not unexpected. Some of diametral expansion in pellets at this swelling level is often compressible because of the development of cracks in the pellet structure. Therefore, the measured pellet swelling data from CRA-548, while not directly used in determining the PCI strain rate, supports the validity of the cladding profilometry measurements.

The PCI strain is also calculated by the CNRD2 code,⁽⁶⁾ based on the B_4C swelling correlation incorporated in the code. Figure F.3-10 shows the predicted PCI strain behavior for the peak pin in both Series 1 and Series 2 absorbers in a Row 5 location. CNRD2 performs calculations in terms of equivalent full-power days (EFPD) at 400-MW operating conditions. Based on fluence calculations performed for the Reload Design Reports (RDRs), 1 EFPD at these conditions is on the average equivalent to a surveillance fluence of 0.0225×10^{22} n/cm² in a Row 5 location and 0.027×10^{22} n/cm² in a Row 3 location. The strain rates shown in Figure F.3-10 are converted to a fluence basis using the Row 5 conversion factor. As indicated, the PCI rate for Series 2 is somewhat lower than for Series 1 (0.30% per 10^{22} n/cm² versus 0.35% per 10^{22} n/cm²). This is because at the relative fluence levels where PCI is first incurred, the B_4C swelling rate (on a fluence basis) for the Series 2 absorber is lower because of a reduced burnup rate in the B_4C from depletion of the boron-10 isotope.

Because the CNRD2 projected PCI strain rate is more conservative than that indicated by the Series 1 PIE data (0.30% versus 0.28% per 10^{22} n/cm²), it was selected to determine the lifetime fluence limit associated with the additional 1.83% allowable strain for the Series 2 absorber. The lifetime fluence limit for Row 5, Series 2 absorbers is thus:

$$\text{Fluence Limit} = 9.9 + 1.83/0.30 - 16.0 \times 10^{22} \text{ n/cm}^2$$

The PCI limits for Series 2 absorbers in Row 3 positions were obtained directly from CNRD2 analyses, which are conservative as indicated above. For a fully withdrawn Row 3 absorber, the surveillance fluence limit is $26.7 \times 10^{22} \text{ n/cm}^2$. If the Row 3 rod were operated continuously in the shimmed configuration (≥ 30 in. withdrawn), the fluence limit is $20.0 \times 10^{22} \text{ n/cm}^2$.

Conservatism inherent in these evaluations include the 1% PCI strain limit (factor of 2 below minimum measured strains in breached pins) and the application of the CNRD2 PCI strain rate (higher than PIE measured indications). Considerable emphasis is placed on the CRA-548 measured PCI tie point (0.17% at $9.9 \times 10^{22} \text{ n/cm}^2$). The PCI measurement uncertainty is 0.12% at the two-sigma level. This would reduce lifetime by $0.4 \times 10^{22} \text{ n/cm}^2$ at the two-sigma uncertainty level, which is substantially less than the margin of $1.0 \times 10^{22} \text{ n/cm}^2$ between the Row 5 nominal PCI lifetime and the Series 2 overall lifetime fluence limit of $15.0 \times 10^{22} \text{ n/cm}^2$.

Internal Gas Pressure--Helium is produced in the neutron capture reaction in B_4C , and some of this helium is released to the absorber pin plenum. The associated gas pressure buildup exerts a load on the cladding and can produce damage and strain in the cladding if high stress levels are reached.

Gas release rates from B_4C are relatively low and are not limiting for the thick-walled Series 2 absorber pin design. CNRD2 calculations were performed out to 1,000 EFPD at 400-MW conditions for both Row 3 and Row 5 locations. This corresponds to surveillance fluence levels of $27 \times 10^{22} \text{ n/cm}^2$ and $23 \times 10^{22} \text{ n/cm}^2$ for Row 3 and Row 5, respectively. At these high exposure levels, the cumulative damage function (CDF) in the cladding from gas pressure loading was zero for both cases. Hoop stress in the Row 5 peak absorber pin at this exposure was 4,166 psi, while in the Row 3 absorber it was only 1,956 psi. These values compare to a yield strength of over 75,000 psi for the cladding at operating temperatures.

An additional conservatism inherent in this evaluation is that the gas release correlation employed in CNRD2 projects approximately twice the gas release quantities as those observed in postirradiation examinations of FFTF absorber pins.

Relative Duct Bowing--Thermal and flux gradients across the absorber assembly cause the inner duct and outer duct assemblies to bow as irradiation proceeds. Should the relative bowing between the two components become severe enough, interference to the vertical translation of the absorber bundle could be encountered. The end-of-life condition for this phenomenon is defined as the point where the relative bowing is such as to cause a forced contact situation between the movable inner duct assembly and the fixed outer duct. This condition corresponds to a configuration where the two wear pads contact the outer duct on one side of the assembly, while the inner duct tube contacts the outer duct on the opposite side of the assembly. This condition is illustrated in an exaggerated sense in Figure F.3-11. Such a situation could impede the vertical travel of the absorber bundle and potentially impact scram response. The full vertical translation (36 in. stroke) must be considered in evaluating this forced contact situation.

Bowing analyses of the Series 2 absorber assembly was performed using the CRPBOW code.⁽⁶⁾ The code performs calculations for steel swelling, irradiation-induced creep, thermal elongation, and beam centerline displacement as a function of operating time and pin bundle axial position. The flux and thermal gradients structured into the code represent FFTF Row 5 control rod conditions at 400-MW power operation. Because these gradients would be less at lower operating power levels (e.g., 291 MW), the evaluations are conservative for these cases (i.e., bowing predictions are exaggerated for equivalent fluence conditions). While the flux and temperature conditions intrinsic in the code are not applicable to Row 3 control rod analyses, the Row 5 lifetime capabilities in terms of fluence limits apply conservatively to Row 3 rods. This is because the flux and temperature gradients are less in a Row 3 location (e.g., the flux decreases about 20% across a Row 5 rod and about 13% across a Row 3 rod). Also, the absorber bundle in a Row 3 rod is normally withdrawn during reactor operation and, therefore, not as subject to bowing tendencies.

To better reflect actual fluence accumulation during each cycle at 291-MW power operation, CRPBOW was run on the basis of 80 EFPD cycles at 400-MW power conditions. As indicated above, this conservatively applies more severe flux

and temperature gradient conditions than actually exist at 291-MW operation, while accruing approximately the correct fluence during each operating cycle at 291-MW. It is assumed that the control rod is steadily withdrawn from 16 in. out to 36 in. out during the course of each cycle.

CRPBOW performs calculations on the basis of EFPD at 400-MW operating conditions. The following factors are used to convert operating time to Row 5 surveillance fluence values: 0.0225×10^{22} n/cm² per EFPD on the side facing core center (Side 1) and 0.0180×10^{22} n/cm² per EFPD on the side facing away from core center (Side 2). These conversion factors were obtained from RDR calculations performed for the FFTF during the period it operated at 400-MW (Cycles 1 through 8).

Analyses were performed for a number of cases where the control rod was rotated at various times (at the end of incremental 80 EFPD cycles). Figure F.3-12 shows typical CRPBOW projections of bowing behavior. In this particular case, the control rod is rotated at 400 EFPD. Side 1 corresponds to the side facing toward core center, with Side 2 representing the opposite side of the absorber assembly. It is seen that when the control rod is rotated, the bowing trends reverse and the clearance on Side 1 begins increasing. Eventually, as bowing continues in this reverse direction, clearance on the opposite side (Side 2) begins decreasing and ultimately becomes limiting if no further rotation is performed. On the basis of the CRPBOW clearance parameter plotted in the figure, a value of -0.100 in. corresponds to the three-point forced contact situation that represents end of life. This condition is reached on Side 2 at an exposure of 710 EFPD. The cumulated surveillance fluence at this point is 14.6×10^{22} n/cm². This value accounts for the reduced fluence accumulation rate on Side 1 after the assembly is rotated. The total fluence on Side 1 remains greatest at this point. The corresponding Side 2 fluence level is 14.2×10^{22} n/cm².

If the assembly is never rotated, the surveillance fluence limit on Side 1 is 11.8×10^{22} n/cm². That is, a three-point forced contact situation is reached at this point because of Side 1 bowing distortions. With rotation, bowing interference on Side 2 eventually becomes limiting (if the assembly is rotated

only once). Figure F.3-13 presents this single-rotation bowing lifetime limit as a function of the surveillance fluence at the time of rotation. The lifetime fluence limits shown account for the reduced fluence accumulation rate on the limiting side after rotation. A straight line fit to the data is included in the figure.

In evaluating the CRPBOW results, it is noted that once the assembly is in the bowing mode (surveillance fluence greater than about $6 \times 10^{22} \text{ n/cm}^2$), the opposite side bowing does not become limiting after rotation until the peak surveillance fluence (side towards core center) exceeds an additional increment of $7 \times 10^{22} \text{ n/cm}^2$. This is true for an rotation fluence of $5 \times 10^{22} \text{ n/cm}^2$ or greater. This is the basis for developing the fluence ranges for the second rotation presented in Table F.3-6. With assembly rotations performed within the specified fluence ranges, the bowing-limited lifetime is at least $15 \times 10^{22} \text{ n/cm}^2$. It is apparent that with appropriate rotation sequences, duct bowing need never become limiting. However, the rotation schedule specified in Table F.3-6 ensures a bowing-limited fluence capability of at least $15 \times 10^{22} \text{ n/cm}^2$.

A number of conservatisms are included in the bowing analyses. Application of 400-MW conditions to 291-MW operation imposes exaggerated flux and temperature gradients across the assembly (i.e., bowing responses would actually be less at lower power conditions). The two-sigma version of the stainless steel swelling correlation (Nuclear Systems Materials Handbook, Rev. 9) was employed to conservatively project bowing behavior. Although Row 3 rod analyses were not specifically performed, the Row 5 results conservatively apply to Row 3 limits with respect to fluence capabilities because of the reduced gradients in Row 3 and the withdrawn mode of operation for the Row 3 absorber bundle.

Pin Bundle-Inner Duct Interaction--Caused primarily by creep strain incurred in the absorber pins, the pin bundle grows faster with irradiation than does the inner duct tube that contains the pin bundle. Thus, the clearance (0.026 in. at fabrication) between the pin bundle and inner duct diminishes with irradiation exposure. Once the pin bundle growth has closed this gap, further pin bundle dilation could cause the inner duct to expand and decrease

the clearance between inner and outer ducts. This, in turn, could impact the bowing limited lifetime. While such pin-bundle dilation could be easily accommodated by appropriate analytical bowing evaluations and control rod rotation schedules, a zero clearance between the pin bundle and inner duct tube is adopted as a conservative limit. Nesting of the pin bundle is ignored for conservatism because the absorber pins are relatively stiff.

CNRD2 calculates the bundle-to-duct clearance as a function of operating exposure. For a Row 5 control rod, the lifetime fluence limit (zero remaining clearance) is $17.8 \times 10^{22} \text{ n/cm}^2$. For a Row 3 rod, the fluence limit is $23.2 \times 10^{22} \text{ n/cm}^2$.

The zero clearance criterion as a lifetime is very conservative, inasmuch as considerable inner duct dilation could be accommodated without impacting bowing limited lifetimes.

F.3.5 OTHER VESSEL INTERNALS

The design of the CLIRAs and the OTAs specifies materials of construction and load pads of matching size and elevation such that the behavior of these in-core vehicles for irradiation testing is equivalent to that of the surrounding fuel assemblies. Expected length increases are accommodated by existing design. For these components, no reduction in the in-core design life of 300 FPD results from operation of the reactor at an inlet temperature of 680 °F.

Other vessel internals adjacent to the core were reviewed, specifically:

- Core Restraint Mechanism
- Instrument Tree
- Control Rod Driveline and Guide Tube
- Reflectors.

New swelling data were considered, where applicable. No shortening of design lifetime results for any of these components for operation at 680 °F. Perhaps the most significant consideration is the potential for increased length of in-core components having long residence. The PIOTA, IVHM, IT, and CRDM accommodate all ducts within design tolerances. However, as a result of this study, acceptance of "over-length" assemblies has been restricted to core positions having lower fluence/burnup ratios or by scheduling early discharge after initial loading.

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TABLE F.3-3
RESULTS OF CORE RESTRAINT ANALYSIS
(Rev. 5 Swelling and Rev. 3 Creep)

<u>Case</u>	<u>Maximum Withdrawal Force** (lbs)</u>	<u>Maximum Bending Stress at EDL + (psi)</u>	<u>Maximum Permanent Simple Bow+ (in.)</u>
nominal R(t), $\tau=5$	2,290*	6,940	0.49
1.3 x R(t), $\tau=5$	4,040*	10,800	0.72
limit criterion	4,000	16,000++	0.90

*Not including assembly weight (~280 lb)

**Row 1

+Row 2

++Margin available for bending stress after considering combined stresses in accordance with Code Case 1952.

TABLE F.3-4

Pellet-Cladding Interaction Experience.

<u>Experiment</u>	<u>Cladding</u>	<u>Clad T (°F)</u>	<u>Peak Strain</u>	<u>Breach</u>
EBR-II/HEDL	304 SST	1,000 - 1,250	0.5%	No
EBR-II/HEDL	316 SST	1,350	1.8%	No
EBR-II/HEDL	316 SST	1,200	1.8%	No
EBR-II/HEDL	316 SST	1,200	1.6%	No
EBR-II/HEDL	316 SST	800	1.3%	No
EBR-II/HEDL	316 SST	800	0.7%	No
FFTF/WHC	316 SST	1,000	1.2%	No
BOR-60/USSR	OX16H15M3B	1,000	0 - 2%	No
BOR-60/USSR	OX16H15M3B	1,000	2 - 4%	Yes

TABLE F.3-5

Data From Series 1 Absorbers Used to
Determine PCI Strain Rate.

<u>Source</u>	<u>Fluence</u>	<u>Pellet ΔD</u>	<u>Clad ID Strain</u>	<u>Clad OD Strain</u>
CRA-548	9.9	4.7%	--	0.17%*
CRA-537	8.2	3.4%	-0.49%	-0.30%
CRA-541	5.6	2.2%	-1.65%	-1.02%
B ₄ C Δ d Onset	1.1	0.0%	-3.76%	-2.33%

*Direct measurement

TABLE F.3-6

Series 1.5 and Series 2 Absorber Assembly Lifetime Limits
at 680 °F Coolant Inlet.

<u>Parameter</u>	<u>Fluence (10^{22} n/cm²)</u>	
	<u>Row 3 Rods*</u>	<u>Row 5 Rods</u>
Bottom Plate-Wear Band Interaction	>15.0	>15.0
Pellet-Cladding Interaction	20.0	16.0
Internal Gas Pressure	>27	>23
Relative Bowing	>15**	>15**
Pin Bundle-Inner Duct Interaction	23.2	17.8

*The values presented for the Row 3 rods reflect operation in the shimmed configuration (≥ 30 in. withdrawn), but conservatively apply to full-out (36 in. withdrawn) conditions as well.

**Requires rotation per following prescribed schedule. Without rotation, the fluence limit is 11.8×10^{22} n/cm².

<u>If First Rotation Performed at:</u>	<u>Then Second Rotation to be Performed at:</u>
$5.0 \leq \phi t < 7.0$ - - - - -	$8.8 \leq \phi t \leq 10.8$
$7.0 \leq \phi t < 9.3$ - - - - -	$10.8 \leq \phi t \leq 12.8$
$9.3 \leq \phi t \leq 11.8$ - - - - -	Not Required

39307041.1

Figure F.3-9. PCI Strain Rate Projected by Series 1 Measurement Data.

Figure F.3-10. CNRD2 Projections of PCI Strain.

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Figure F.3-11. End-of-Life Condition from Bowing Interference.

39307041.4

Figure F.3-12. Bowing Behavior with Rotation at 400 EFPD.

39307041.5

Figure F.3-13. Bowing Lifetime Limit with Single Rotation.

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FFTF Series I (Cores 1 and 2) and Series II (Cores 3 and 4) driver fuel assemblies. In doing so, it is necessary to ensure that the actual operating conditions of an assembly (absolute power, power/flow ratio, and power history) are bounded by the assumptions of the fuel lifetime analyses from which the curves of Figure F.A-2 are derived. This can be ensured through the appropriate choice of an equivalent BOL power according to the following method:

- (1) The actual power history and projected power history through the intended operating life should be examined and an equivalent BOL power level chosen such that the actual power history is bounded by a decreasing power ramp, beginning at the equivalent BOL power and decreasing by 15.5% at a peak burnup of 80,000 MWd/MTM.
- (2) The ratio of the equivalent BOL power, divided by the assembly flow that was assumed for the analysis (shown in Figure F.A-2), must equal or exceed the nominal calculated power/flow ratio for the fuel assembly, using the equivalent BOL power and the minimum flow throughout its irradiation lifetime.

In cases that are poorly approximated by these conservative assumptions, an explicit fuel lifetime calculation may be made using actual operating history and following the method of Reference F.A.2. The procedure used to implement the Cumulative Damage Fraction (CDF) method developed in Reference F.A.2 for the driver fuel lifetime analyses, including the method of entering data in the calculational tool, is described in more detail in Reference F.A.4. The calculational tool was updated to the LIFE-4, Revision 1 code, from a developmental version of the LIFE-4, Revision 0 code, used in the original work described in Reference F.A.2. The temperature dependent CDF lifetime criteria obtained using the Life-4, Revision 1 code, is similar to that previously obtained for the Life-4, Revision 0 code, but is generally more conservative. The CDF criteria developed with the LIFE-4, Revision 1 code, was validated against the same data set as the LIFE-4, Revision 0 code, (eight EBR-II breached pins) plus six FFTF pins (five breached pins).

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