UNITED STATES

ATOMIC ENERGY COMMISSION

WASHINGTON, DC 20545

BOSTON EDISON COMPANY

(PILGRIM NUCLEAR POWER STATION)

DOCKET NO. 50-293

FACILITY OPERATING LICENSE

License No. DPR-35

The Atomic Energy Commission (the Commission) having found that:

a. Except as stated in condition 5, construction of the Pilgrim Nuclear Power Station (the facility) has been substantially completed in conformity with the application, as amended, the Provisional Construction Permit No. CPPR-49, the provisions of the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission as set forth in Title 10, Chapter 1, CFR; and

b. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and

c. There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission; and

d. The Boston Edison Company (Boston Edison) is technically and financially qualified to engage in the activities authorized by this operating license, in accordance with the rules and regulations of the Commission; and

e. Boston Edison has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations; and

f. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public; and

g. In accordance with the requirements of Appendix D to 10 CFR Part 50, the operating license should be issued subject to conditions for protection of the environment set forth herein.

Facility Operating License No. DPR-35, dated June 8, 1972, issued to the Boston Edison Company (Boston Edison) is hereby amended in its entirety, pursuant to an Initial Decision dated September 13, 1972, by the Atomic Safety and Licensing Board, to read as follows:

1. This license applies to the Pilgrim Nuclear Power Station, a single cycle, forced circulation, boiling water nuclear reactor and associated electric generating equipment (the facility). The facility is located on the western shore of Cape Cod Bay in the town of Plymouth on the Boston Edison site in Plymouth County, Massachusetts, and is described in the "Final Safety Analysis Report," as supplemented and amended.

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Boston Edison:

   A. Pursuant to the Section 104b of the Atomic Energy Act of 1954, as amended (the Act) and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location on the Pilgrim site;

   B. Pursuant to the Act and 10 CFR 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

   C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

   D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

   E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50 and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. <u>Maximum Power Level</u>

      Boston Edison is authorized to operate the facility at steady state power levels not to exceed 1998 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised by the NRC approval and docketed Amendents, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Records

Boston Edison shall keep facility operating records in accordance with the requirements of the Technical Specifications.

D. Equalizer Valve Restriction - DELETED

E. Recirculation Loop Inoperable

The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

F. Fire Protection

Boston Edison shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated December 21, 1978 as supplemented subject to the following provision:

Boston Edison may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

G. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10CFR73.55 (51FR27817 and 27822) and to the authority of 10CFR50.90 and 10CFR50.54(p). The plans, which contain Safeguards Information protected under 10CFR73.21, are entitled: "Pilgrim Nuclear Power Station Physical Security Plan," with revisions submitted through September 18, 1987; "Pilgrim Nuclear Power Station Guard Training and Qualification Plan," with revisions submitted through September 24, 1984; and "Pilgrim Nuclear Power Station Safeguards Contingency Plan," with revisions submitted through February 15, 1984. Changes made in accordance with 10CFR73.55 shall be implemented in accordance with the schedule set forth therein.

H. Long Term Program

(1) The "Plan for the Long Term Program for Pilgrim Nuclear Power Station" (the Plan) submitted on May 7, 1984, is approved.

a) The Plan shall be followed by the licensee from and after the effective date of this amendment.

b) Changes to dates for completion of items identified in Schedule B of the Plan do not require a license amendment. Dates specified in Schedule A shall be changed only in accordance with applicable NRC procedure.

I. Post-Accident Sampling System, NUREG-0737, Item II.B.3, and Containment Atmospheric Monitoring System, NUREG-0737, Item II.F.1(6)

The licensee shall complete the installation of a post-accident sampling system and a containment atmospheric monitoring system as soon as practicable, but no later than June 30, 1985.

4. This license is subject to the following condition for the protection of the environment: Boston Edison shall continue, for a period of five years after initial power operation of the facility, an environmental monitoring program similar to that presently existing with the Commonwealth of Massachusetts (and described generally in Section C-III of Boston Edison's Environmental Report, Operating License Stage dated September, 1970) as a basis for determining the extent of station influence on marine resources and shall mitigate adverse effects, if any, on marine resources.

5. Boston Edison has not completed as yet construction of the Rad Waste Solidification System and the Augmented Off-Gas System. Limiting conditions concerning these systems are set forth in the Technical Specifications.

6. Pursuant to Section 105c(8) of the Act, the Commission has consulted with the Attorney General regarding the issuance of this operating license. After said consultation, the Commission has determined that the issuance of this license, subject to the conditions set forth in this subparagraph 6., in advance of consideration of and findings with respect to matters covered in Section 105c of the Act, is necessary in the public interest to avoid unnecessary delay in the operation of the facility. At the time this operating license is being issued an antitrust proceeding has not been noticed. The Commission, accordingly, has made no determination with respect to matters covered in Section 105c of the Act, including conditions, if any, which may be appropriate as a result of the outcome of any antitrust proceeding. On the basis of its findings made as a result of an antitrust proceeding, the Commission may continue this license as issued, rescind this license or amend this license to include such conditions as the Commission deems appropriate. Boston Edison and others who may be affected hereby are accordingly on notice that the granting of this license is without prejudice to any subsequent licensing action, including the imposition of appropriate conditions, which may be taken by the Commission as a result of the outcome of any antitrust proceeding. In the course of its planning and other activities, Boston Edison will be expected to conduct itself accordingly.

7. This license is effective as of the date of issuance and shall expire June 8, 2012.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by A. Giambusso

A. Giambusso, Deputy Director for Reactor Projects
Directorate of Licensing

Attachments:
Appendix A - Technical Specifications (Radiological)

Date of Issuance: September 15, 1972

APPENDIX A

TO

FACILITY OPERATING LICENSE DPR-35

TECHNICAL SPECIFICATION AND BASES

FOR

PILGRIM NUCLEAR POWER STATION

PLYMOUTH, MASSACHUSETTS

BOSTON EDISON COMPANY

DOCKET NO. 50-293

Dupe

TECHNICAL SPECIFICATIONS

PILGRIM NUCLEAR POWER STATION

Boston Edison

PNPS TECHNICAL SPECIFICATIONS

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1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

B. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

D. Core Operating Limits Report

The CORE OPERATING LIMITS REPORT is a reload-cycle specific document, its supplements and revisions, that provides core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.A.4. Plant operation within these operating limits is addressed in individual specifications.

E. Operable - Operability

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

H. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% design power.

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 600 psig, the main steam isolation valves closed and the mode switch in startup.

J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

K. Mode - The reactor mode is that which is established by the mode-selector-switch. The modes include shutdown, refuel, startup and run which are defined as follows:

1. Startup Mode - In this mode the reactor protection scram trip, initiated by main steam line isolation valve closure, is bypassed when reactor pressure is less than 600 psig, the low pressure main steam line isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

2. Run Mode - In this mode the reactor system pressure is at or above 785 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

   a. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

   b. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

4. Refuel Mode - The reactor is in the refuel mode when the mode switch is in the refuel mode position. When the mode switch is in the refuel position, the refueling interlocks are in service.

L. Design Power - Design power means a steady-state power level of 1998 thermal megawatts.

M. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All blind flanges and manways are closed.

4. All automatic primary containment isolation valves are operable or at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition.

5. All containment isolation check valves are operable or at least one containment valve in each line having an inoperable valve is secured in the isolated position.

N. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

O. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

P. Refueling Frequencies:

1. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 11 months of completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage. (Definitions U and V apply)

2. Refueling Interval - Refueling interval applies only to ASME Code, Section XI IWP and IWV surveillance tests. For the purpose of designating frequency of these code tests, a refueling interval shall mean at least once every 24 months.

Q. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

R. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

S. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - the value of critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3. Total Peaking Factor - The ratio of the fuel rod surface heat flux to the heat flux of an average rod in an identical geometry fuel assembly operating at the core average bundle power.

T. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm or trip.

2. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

4. Instrument Check - An instrument check is a qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

T. Instrumentation (Cont)

5. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves opened.

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

7. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

8. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

9. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

U. Surveillance Frequency - Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

The Surveillance Frequency establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance schedule and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Definition "U" is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

V. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable. The operating cycle interval is 24 months and the 25% tolerance given in Definition "U" is applicable. The refueling interval is 24 months and the 25% tolerance specified in definition "U" is applicable.

W. Fire Suppression Water System - A fire suppression water system shall consist of: a water source(s); gravity tank(s) or pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include hydrant post indicator valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

X. Staggered Test Basis - A staggered test basis shall consist of: (a) a test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals; (b) the testing of one system, subsystem, train or other designated components at the beginning of each subinterval.

Y. Source Check - A source check shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

Z. Offsite Dose Calculation Manual (ODCM) - An offsite dose calculation manual (ODCM) shall be a manual containing the current methodology and parameters to be used for the calculation of offsite doses due to radioactive gaseous and liquid effluents, the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints, and the conduct of the Radiological Environmental Monitoring Program.

AA. Action - Action shall be that part of a specification which prescribes remedial measures required under designated conditions.

BB. Member(s) of the Public[1] - Member(s) of the public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the site.

CC. Site Boundary - The site boundary is shown in Figure 1.6-1 in the FSAR.

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[1] See FSAR Figure 1.6-1

DD. Radwaste Treatment System

1. Gaseous Radwaste Treatment System - The gaseous radwaste treatment system is that system identified in Figure 4.8-2.

2. Liquid Radwaste Treatment System - The liquid radwaste treatment system is that system identified in Figure 4.8-1.

EE. Automatic Primary Containment Isolation Valves - Are primary containment isolation valves which receive an automatic primary containment group isolation signal.

FF. Pressure Boundary Leakage - Pressure boundary leakage shall be leakage through a non-isolable fault in a reactor coolant system component body, pipewall or vessel wall.

GG. Identified Leakage - Identified leakage shall be:

1. Reactor coolant leakage into drywell collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

2. Reactor coolant leakage into the drywell atmosphere from sources which are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be Pressure Boundary Leakage.

HH. Unidentified Leakage - Unidentified leakage shall be all reactor coolant leakage which is not Identified Leakage.

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.07.

2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 Safety Limit Violation

With any Safety Limit not met the following actions shall be met:

2.2.1 Within one hour notify the NRC Operations Center in accordance with 10CFR50.72.

2.2.2 Within two hours:

A. Restore compliance with all Safety Limits, and

B. Insert all insertable control rods.

2.2.3 The Station Director and Senior Vice President - Nuclear and the Nuclear Safety Review and Audit Committee (NSRAC) shall be notified within 24 hours.

2.2.4 A Licensee Event Report shall be prepared pursuant to 10CFR50.73. The Licensee Event Report shall be submitted to the Commission, the Operations Review Committee (ORC), the NSRAC and the Station Director and Senior Vice President - Nuclear within 30 days of the violation.

2.2.5 Critical operation of the unit shall not be resumed until authorized by the Commission.

2.0 SAFETY LIMITS

INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish a Safety Limit such that the Minimum Critical Power Ratio (MCPR) is not less than the limit specified in Specification 2.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling (i.e., MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling.

FUEL CLADDING INTEGRITY (2.1.1)

GE critical power correlations are applicable for all critical power calculations at pressures at or above 785 psig or core flows at or above 10% of rated flow. For operation at low pressures and low flows another basis is used as follows:

(Cont)

FUEL CLADDING INTEGRITY (2.1.1) (Cont)

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (1), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psig	
Max Flux:	0.1 to 1.25x10^6 lb/hr-ft^2	
Inlet Subcooling:	0 to 100 Btu/lb	
Local Peaking:	1.61 at a corner rod to	
	1.47 at an interior rod	
Axial Peaking:	Shape	Max/Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array	16,64 Rods in an 8x8 array	
	49 Rods in an 7x7 array	

MINIMUM CRITICAL POWER RATIO (2.1.2)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not result in damage to BWR fuel rods, the critical power at

(Cont)

MINIMUM CRITICAL POWER RATIO (2.1.2) (Cont)

which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity Safety Limit calculation are given in Reference 1. Reference 1 includes a tabulation of the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of the parameters used in the Safety Limit MCPR statistical analysis.

The statistical analysis used to determine the MCPR safety limit is based on a model of the BWR core which simulates the process computer function. The reactor core selected for these analyses was a large 764 assembly, 251 inch reload core. Results from the large reload core analysis apply for all operating reactors for all reload cycles, including equilibrium cycles. Random Monte Carlo selections of all operating parameters based on the uncertainty ranges of manufacturing tolerances, uncertainties in measurement of core operating parameters, calculational uncertainties, and statistical uncertainty associated with the critical power correlations are imposed upon the analytical representation of the core and the resulting bundle critical power ratios. Details of this statistical analysis are presented in Reference 2.

REACTOR WATER LEVEL (Shutdown Condition) (2.1.3)

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored.

(Cont)

REACTOR STEAM DOME PRESSURE (2.1.4)

The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition, including the January 1966 Addendum), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the USAS Nuclear Power Piping Code, Section B31.1.0 for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig at 562°F for suction piping and 1241 psig at 562°F for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

REFERENCES

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (Applicable Amendment specified in the CORE OPERATING LIMITS REPORT).

2. General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, January 1977, NEDE-10958-PA and NEDO-10958-A.

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective:

To assure the operability of the reactor protection system.

Specification:

The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milli-seconds.

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification:

Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2 respectively.

PNPS Table 3.1.1 REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Operable Inst. Channels per Trip System (1) Minimum	Avail.	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Refuel (7)	Startup/Hot Standby	Run	Action (1)
1	1	Mode Switch in Shutdown		X	X	X	A
1	1	Manual Scram		X	X	X	A
		IRM					
3	4	High Flux	≤120/125 of full scale	X	X	(5)	A
3	4	Inoperative		X	X	(5)	A
		APRM					
2	3	High Flux	(15)	(17)	(17)	X	A or B
2	3	Inoperative	(13)	X	X(9)	X	A or B
2	3	High Flux (15%)	≤15% of Design Power	X	X	(16)	A or B
2	2	High Reactor Pressure	≤1063.5 psig	X(10)	X	X	A
2	2	High Drywell Pressure	≤2.22 psig	X(8)	X(8)	X	A
2	2	Reactor Low Water Level	≥11.7 In. Indicated Level	X	X	X	A
		SDIV High Water Level:	≤38 Gallons	X(2)	X	X	A
2	2	East					
2	2	West					
4	4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	A or C
2	2	Turbine Control Valve Fast Closure	≥150 psig Control Oil Pressure at Acceleration Relay	X(4)	X(4)	X(4)	A or D
4	4	Turbine Stop Valve Closure	≤10% Valve Closure	X(4)	X(4)	X(4)	A or D

NOTES FOR TABLE 3.1.1

1. There shall be two operable or tripped trip systems for each trip function (e.g., high drywell pressure, reactor low water level, etc.). An instrument channel, satisfying minimum operability requirements for a trip system, may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

   An inoperable channel and/or trip system need not be placed in the tripped condition if this would cause a full scram to occur. When a trip system can be placed in the tripped condition without causing a full scram to occur, place the trip system with the most inoperable channels in the tripped condition, per the table below. If both systems have the same number of inoperable channels, place either trip system in the tripped condition, per the table below.

Condition	Required Action	Completion Time
a. With less than the minimum required operable channels per trip function in one trip system.	Place associated trip system in trip or *	12 hours
b. With less than the minimum required operable channels per trip function, in both trip systems.	Place one trip system in trip or *	6 hours
c. If full scram trip capability is not available for a given trip function	Restore RPS trip capability or *	1 hour

   * Initiate the actions required by Table 3.1.1 and specified in Actions A through D below for that function:

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four (4) hours.

   B. Reduce power level to IRM range and place mode switch in the startup/hot standby position within eight (8) hours.

   C. Reduce turbine load and close main steam line isolation valves within eight (8) hours.

   D. Reduce power to less than 45% of design.

NOTES FOR TABLE 3.1.1 (Cont)

2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.
3. Permissible to bypass when reactor pressure is <576 psig.
4. Permissible to bypass when turbine first stage pressure is less than ≤ 112 psig.
5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.
6. The design permits closure of any two lines without a scram being initiated.
7. When the reactor is subcritical, fuel is in the reactor vessel and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   A. Mode switch in shutdown
   B. Manual [illegible]
   C. High [illegible]
   D. Scram [illegible] ge volume high level
   E. APRM ([illegible] high flux scram

8. Not required to be operable when primary containment integrity is not required.
9. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).
10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.
11. Deleted
12. Deleted
13. An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there is less than 50% of the normal complement of LPRM's to an APRM.
14. Deleted
15. The APRM high flux trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT, but shall in no case exceed 120% of rated thermal power.
16. The APRM (15%) high flux scram is bypassed when in the run mode.
17. The APRM flow biased high flux scram is bypassed when in the refuel or startup/hot standby modes.
18. Deleted.

PNPS TABLE 4.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENTATION AND CONTROL CIRCUITS

	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	Trip Channel and Alarm	Every 3 Months
RPS Channel Test Switch (5)	Trip Channel and Alarm	Once per week
IRM		
High Flux	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
Inoperative	Trip Channel and Alarm	Once Per Week During Refueling and Before Each Startup
APRM		
High Flux	Trip Output Relays (4)	Every 3 Months (7)
Inoperative	Trip Output Relays (4)	Every 3 Months
Flow Bias	Trip Output Relays (4)	Every 3 Months
High Flux (15%)	Trip Output Relays (4)	Once Per Week During Refueling and Before Each Startup
High Reactor Pressure	Trip Channel and Alarm (4)	Every 3 Months
High Drywell Pressure	Trip Channel and Alarm (4)	Every 3 Months
Reactor Low Water Level	Trip Channel and Alarm (4)	Every 3 Months
High Water Level in Scram Discharge Tanks	Trip Channel and Alarm (4)	Every 3 Months
Main Steam Line Isolation Valve Closure	Trip Channel and Alarm	Every 3 Months
Turbine Control Valve Fast Closure	Trip Channel and Alarm	Every 3 Months
Turbine First Stage Pressure Permissive	Trip Channel and Alarm (4)	Every 3 Months
Turbine Stop Valve Closure	Trip Channel and Alarm	Every 3 Months
Reactor Pressure Permissive	Trip Channel and Alarm (4)	Every 3 Months

NOTES FOR TABLE 4.1.1

1. Deleted

2. Deleted

3. Functional tests are not required when the systems are not required to be operable or are tripped.

   If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. Test RPS channel after maintenance.

6. Deleted

7. This APRM testing will be performed once every 3 months when in the RUN mode and within 24 hours after entering RUN mode, if not performed within the previous seven days.

PNPS TABLE 4.1.2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Calibration Test (5)	Minimum Frequency (2)
IRM High Flux	Comparison to APRM on Controlled Shutdowns	Note (4)
	Full Calibration	Once per Operating Cycle
APRM High Flux		
Output Signal	Heat Balance	Once every 3 Days
Flow Bias Signal	Calibrate Flow Comparator and Flow Bias Network	At least once every 18 Months
	Calibrate Flow Bias Signal (1)	Every 3 Months
LPRM Signal	TIP System Traverse	Every 1000 Effective Full Power Hours
High Reactor Pressure	Note (7)	Note (7)
High Drywell Pressure	Note (7)	Note (7)
Reactor Low Water Level	Note (7)	Note (7)
High Water Level in Scram Discharge Tanks	Note (7)	Note (7)
Main Steam Line Isolation Valve Closure	Note (6)	Note (6)
Turbine First Stage Pressure Permissive	Note (7)	Note (7)
Turbine Control Valve Fast Closure	Standard Pressure Source	Every 3 Months
Turbine Stop Valve Closure	Note (6)	Note (6)
Reactor Pressure Permissive	Note (7)	Note (7)

NOTES FOR TABLE 4.1.2

1. Adjust the flow bias trip reference, as necessary, to conform to a calibrated flow signal.

2. Calibration tests are not required when the systems are not required to be operable or are tripped.

3. Deleted.

4. Maximum frequency required is once per week.

5. Response time is not a part of the routine instrument channel test, but will be checked once per operating cycle.

6. Physical inspection and actuation of these position switches will be performed during the refueling outages.

7. Calibration of these devices will be performed during refueling outages.

   To verify transmitter output, a daily instrument check will be performed. Calibration of the associated analog trip units will be performed concurrent with functional testing as specified in Table 4.1.1.

BASES:

3.1 REACTOR PROTECTION SYSTEM

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (Reference FSAR Section 7.2). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic (i.e., an input signal on either one or both of the subchannels will cause a trip system trip). The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure, and the Turbine Stop Valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved (i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor). Three APRM instrument channels are provided for each protection trip system.

For some trip functions (e.g. MSIV or Turbine Stop Valve Position Switches), the loss of one instrument may lead to degradation of both trip systems. In these cases, a 6 hour LCO must be entered.

A source range monitor (SRM) system is also provided to supply additional neutron level information during refuel and startup (Reference FSAR Section 7.5.4).

3.1 REACTOR PROTECTION SYSTEM (Cont)

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale assures there is proper overlap in the neutron monitoring systems and thus, sufficient coverage is provided for all ranges of reactor operation.

The provision of an APRM scram at ≤15% design power in the Refuel and Startup/Hot Standby modes and the backup IRM scram at ≤120/125 of full scale assures there is proper overlap in the Neutron Monitoring Systems and thus, sufficient coverage is provided for all ranges of reactor operation.

The APRM's cover the Refuel and Startup/Hot Standby modes with the APRM 15% scram, and the power range with the flow-biased rod block and scram. The IRM's provide additional protection in the Refuel and Startup/Hot Standby modes. Thus, the IRM and APRM 15% scram are required in the Refuel and Startup/Hot Standby modes. In the power range, the APRM system provides the required protection (Reference FSAR Section 7.5.7). Thus, the IRM system is not required in the Run mode.

The high reactor pressure, high drywell pressure, reactor low water level, and scram discharge volume high level scrams are required for Startup/Hot Standby and Run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions, as indicated in Table 3.1.1, operable in the Refuel mode is to assure shifting to the Refuel mode during reactor power operation does not diminish the capability of the reactor protection system.

Below 176 psig (analytical limit) turbine first-stage pressure (45% of rated core thermal power for the most limiting balance-of-plant configuration), the scram signals due to turbine stop valve closure or fast closure of turbine control valves are bypassed because flux and pressure scram are adequate to protect the reactor. If the scram signal due to turbine stop valve closure or fast closure of turbine control valves is bypassed at lower powers, less conservative MCPR and MAPLHGR operating limits may be applied as specified in the CORE OPERATING LIMITS REPORT.

Average Power Range Monitor (APRM)

APRM's #1 and #3 operate contacts in one subchannel and APRM's #2 and #3 operate contacts in the other subchannel. APRM's #4, #5, and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel.

3.1 REACTOR PROTECTION SYSTEM (Cont)

The APRM system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of design power (1998 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrated that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage. Therefore, the use of flow-referenced scram trip provides even additional margin.

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses.

Thus, the APRM setting was selected because it provides proper margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

Analyses of the limiting transients show that no scram adjustment is required to assure the minimum critical power ratio (MCPR) is greater than the safety limit MCPR when the transient is initiated from MCPR above the operating limit MCPR.

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides proper thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is sufficient to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer.

Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable case of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally the heat flux is in the near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before power

could exceed the safety limit. The 15% APRM scram remains active until the mode switch is placed in the RUN position.

The analysis to support operation at various power and flow relationships has considered operation with two recirculation pumps.

Intermediate Range Monitor (IRM)

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size.

The IRM scram setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on Range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux, and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded.

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak core power limited to one percent of rated power, thus maintaining MCPR above the safety limit MCPR. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

Reactor Low Water Level

The setpoint for low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results show that scram at this level properly protects the fuel and the pressure barrier, because MCPR remains well above the safety limit MCPR in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 15 inches below the normal operating range and is thus sufficient to avoid spurious scrams.

3.1 REACTOR PROTECTION SYSTEM (Cont)

Turbine Stop Valve Closure

The turbine stop valve closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the safety limit MCPR even during the worst case transient that assumes the turbine bypass is closed.

Turbine Control Valve Fast Closure

The turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the bypass valves. The reactor protection system initiates a scram when fast closure of the control valves is initiated by the acceleration relay. This setting and the fact that control valve closure time is approximately twice as long as that for the stop valves means that resulting transients, while similar, are less severe than for stop valve closure. MCPR remains above the safety limit MCPR.

Main Steam Line Isolation Valve Closure

The low pressure isolation of the main steam lines at 810 psig (as specified in Table 3.2.A) was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 785 psig requires the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram and APRM 15% scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

3.1 REACTOR PROTECTION SYSTEM (Cont)

High Reactor Pressure

The high reactor pressure scram setting is chosen slightly above the maximum normal operating pressure to permit normal operation without spurious scram, yet provide a wide margin to the ASME Section III allowable reactor coolant system pressure (1250 psig, see Bases Section 3.6.D).

High Drywell Pressure

Instrumentation for the drywell is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the Core Standby Cooling Systems (CSCS) initiation to minimize the energy that must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

Reactor Mode Switch

The reactor mode switch actuates or bypasses the various scram functions appropriate to the particular plant operating status (Reference FSAR Section 7.2.3.9).

Manual Scram

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

Scram Discharge Instrument Volume

The control rod drive scram system is designed so that all of the water that is discharged from the reactor by a scram can be accommodated in the discharge piping. The two scram discharge volumes have a capacity of 48 gallons of water each and are at the low points of the scram discharge piping.

3.1 REACTOR PROTECTION SYSTEM (Cont)

During normal operation the scram discharge volume system is empty; however, should it fill with water, the water discharged to the piping could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, redundant and diverse level detection devices in the scram discharge instrument volumes have been provided. The instruments are set to alarm, initiate a control rod block and scram the reactor at three different progressively increasing water levels in the volume. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function properly.

4.1 REACTOR PROTECTION SYSTEM

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with General Electric Company Topical Report NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System," as approved by the NRC and documented in the safety evaluation report (NRC letter to T. A. Pickens from A. Thadani dated July 15, 1987).

A comparison of Tables 4.1.1 and 4.1.2 indicates that two instrument channels have not been included in the latter table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable (i.e., the switch is either on or off).

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating every three days using heat balance data and by calibrating individual LPRM's every 1000 effective full power hours using TIP traverse data.

LIMITING CONDITION FOR OPERATION

3.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the plant instrumentation which initiates and controls a protective function.

Objective:

To assure the operability of protective instrumentation.

Specifications:

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2.A.

B. Core and Containment Cooling Systems - Initiation & Control

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2.B. This instrumentation must be operable when the system(s) it initiates or controls are required to be operable as specified in Section 3.5.

SURVEILLANCE REQUIREMENT

4.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the surveillance requirement of the instrumentation that initiates and controls protective function.

Objective:

To specify the type and frequency of surveillance to be applied to protective instrumentation.

Specifications:

A. Primary Containment Isolation Functions

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.A.

System logic shall be functionally tested as indicated in Table 4.2.A.

B. Core and Containment Cooling Systems - Initiation & Control

Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.B.

System logic shall be functionally tested as indicated in Table 4.2.B.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.2 PROTECTIVE INSTRUMENTATION (Cont)	4.2 PROTECTIVE INSTRUMENTATION (Cont)
C. Control Rod Block Actuation	C. Control Rod Block Actuation
1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.C-1. The trip setpoints for this instrumentation shall be set consistent with Table 3.2.C-2.	1. Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.C. System logic shall be functionally tested as indicated in Table 4.2.C.
D. Radiation Monitoring Systems - Isolation & Initiation Functions	D. Radiation Monitoring Systems - Isolation & Initiation Functions
1. Reactor Building Isolation and Control System and Standby Gas Treatment System The limiting conditions for operation are given in Table 3.2.D.	1. Reactor Building Isolation and Control System and Standby Gas Treatment System Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D. System logic shall be functionally tested as indicated in Table 4.2.D.

LIMITING CONDITION FOR OPERATION

3.2 PROTECTIVE INSTRUMENTATION (Cont)

E. Drywell Leak Detection

The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Section 3.6.C.

F. Surveillance Information Readouts

The limiting conditions for the instrumentation that provides surveillance information readouts are given in Table 3.2.F.

SURVEILLANCE REQUIREMENT

4.2 PROTECTIVE INSTRUMENTATION (Cont)

E. Drywell Leak Detection

Instrumentation shall be functionally tested, calibrated and checked as indicated in Section 4.6.C.

F. Surveillance Information Readouts

Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

LIMITING CONDITION FOR OPERATION

3.2 PROTECTIVE INSTRUMENTATION (Cont)

G. Recirculation Pump Trip/Alternate Rod Insertion Initiation.

This system is only required when the reactor mode switch is in the RUN mode.

The recirculation pump trip system causes a pump trip and the alternate rod insertion system provides for initiating control rod insertion on a signal of high reactor pressure or low-low reactor water level when the mode select switch is in the RUN mode.

The limiting conditions for operation for the instrumentation are listed in Table 3.2-G.

SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENTATION (Cont)

G. Recirculation Pump Trip/Alternate Rod Insertion

Surveillance for instrumentation which initiates Recirculation Pump Trip and Alternate Rod Insertion shall be specified in Table 4.2-G.

3.2 PROTECTIVE INSTRUMENTATION (Cont)

H. Drywell Temperature

1. The drywell temperature shall be maintained within the following limits when the reactor coolant temperature is above 212°F.

   Above elevation 40': ≤ 194°F

   Equal to or Below elevation

   40': ≤ 150°F

   Upon determination that the drywell temperature at any elevation has exceeded the above limits, the drywell temperature at each elevation shall be logged every 30 minutes. The drywell temperature shall be reduced to within the limits within 24 hours; otherwise, corrective action shall be as specified in 3.2.H.2, 3.2.H.3.

2. If the drywell temperature has exceeded either limit of 3.2.H.1 for greater than 24 hours, an engineering evaluation shall immediately be initiated to assess potential damage and render a determination of ability of safety related equipment to perform its intended function.

   If either limit of section 3.2.H.1 has been exceeded for greater than 24 hours, further action to justify continued operation shall be determined by an engineering evaluation, which must be completed within one week.

3. If the requirements of 3.2.H.2 have not been met an orderly Shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

4.2 PROTECTIVE INSTRUMENTATION (Cont)

H. Drywell Temperature

1. When reactor coolant temperature is above 212°F, the drywell air temperature limits will be determined by reading the instruments listed in Table 3.2.H. These instruments shall be logged once per shift, and each reading compared to the limits of Section 3.2.H.1.

2. Instrumentation shall be calibrated and checked as indicated in Table 4.2.H.

3.2 PROTECTIVE INSTRUMENTATION (Cont)

4. If the drywell temperature at any elevation exceeds 215°F and the temperature cannot be reduced to below 215°F within 30 minutes a reactor shutdown shall be initiated and the reactor shall be in cold shutdown condition within 24 hours.

5. The limiting conditions of operation for the instrumentation that monitors drywell temperature are given in Table 3.2.H.

4.2 PROTECTIVE INSTRUMENTATION (Cont)

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Operable Instrument Channels Per Trip System (1)				
Minimum	Available	Instrument	Trip Level Setting	Action (2)
2(7)	2	Reactor Low Water Level	≥11.7" indicated level (3)	A and D
1	1	Reactor High Pressure	≤110 psig	D
2	2	Reactor Low-Low Water Level	at or above - 46.3 in. indicated level (4)	A
2	2	Reactor High Water Level	≤45.3" indicated level (5)	B
2(7)	2	High Drywell Pressure	≤2.22 psig	A
2	2	Low Pressure Main Steam Line	≥810 psig (8)	B
2(6)	2	High Flow Main Steam Line	≤136% of rated steam flow	B
2	2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤170°F	B
2	2	Turbine Basement Exhaust Duct High Temperature	≤150°F	B
1	1	Reactor Cleanup System High Flow	≤300% of rated flow	C
2	2	Reactor Cleanup System High Temperature	≤150°F	C

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function. An instrument channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter; or, where only one channel exists per trip system, the other trip system shall be operable.

2. Action

   If the minimum number of operable instrument channels cannot be met for one of the trip systems of a trip function, the appropriate conditions listed below shall be followed:

   If placing the inoperable channel(s) in the tripped condition would not cause an isolation, the inoperable channel(s) and/or that trip system shall be placed in the tripped condition within one hour (twelve hours for Reactor Low Water Level and High Drywell Pressure) or initiate the action required by Table 3.2.A for the affected trip functions.

   If placing the inoperable channel(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to operable status within two hours (six hours for Reactor Low Water Level and High Drywell Pressure) or initiate the Action required by Table 3.2.A for the affected trip function.

   If the minimum number of operable instrument channels cannot be met for both trip systems, place at least one trip system (with the most inoperable channels) in the tripped condition within one hour or initiate the appropriate Action required by Table 3.2.A listed below for the affected trip function.

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate Shutdown Cooling.

NOTES FOR TABLE 3.2.A (Cont)

3. Instrument set point corresponds to 128.26 inches above top of active fuel.

4. Instrument set point corresponds to 77.26 inches above top of active fuel.

5. Not required in Run Mode (bypassed by Mode Switch).

6. Two required for each steam line.

7. These signals also start SBGTS and initiate secondary containment isolation.

8. Only required in Run Mode (interlocked with Mode Switch).

9. Deleted.

TABLE 3.2.B

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Reactor Low-Low Water Level	at or above -46.3 in. indicated level (4)	1. In conjunction with Low Reactor Pressure, initiates Core Spray and LPCI. 2. In conjunction with High Drywell Pressure, 94.4-115.6 second time delay and LPCI or Core Spray pump interlock initiates Auto Blowdown (ADS). 3. Initiates HPCI; RCIC. 4. Initiates starting of Diesel Generators.
2	Reactor High Water Level	≤+45.3" indicated level	Trips HPCI and RCIC turbines.
1	Reactor Low Level (inside shroud)	>-151" indicated level	Prevents inadvertent operation of containment spray during accident condition. (Indication of 2/3 core coverage.)
2	Containment High Pressure	$1.55 \leq p \leq 1.82$ psig	Prevents inadvertent operation of containment spray during accident condition. Instrument is set to trip at or before 1.82 increasing and reset at or before 1.55 decreasing.

TABLE 3.2.B (Cont)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	High Drywell Pressure	≤2.22 psig	1. Initiates Core Spray; LPCI; HPCI. 2. In conjunction with Low-Low Reactor Water Level, 94.4-115.6 second time delay and LPCI or Core Spray pump running, initiates Auto Blowdown (ADS) 3. Initiates starting of Diesel Generators 4. In conjunction with Reactor Low Pressure initiates closure of HPCI vacuum breaker containment isolation valves.
1	Reactor Low Pressure	400 psig ± 5	Permissive for opening Core Spray and LPCI Admission valves.
1	Reactor Low Pressure	≤110 psig	In conjunction with PCIS signal permits closure of RHR (LPCI) injection valves.
1	Reactor Low Pressure	400 psig ± 5	In conjunction with Low-Low Reactor Water Level initiates Core Spray and LPCI.
2	Reactor Low Pressure	900 psig ± 5	Prevents actuation of LPCI break detection circuit.
2	Reactor Low Pressure	80 psig ± 5	Isolates HPCI and in conjunction with High Drywell Pressure initiates closure of HPCI vacuum breaker containment isolation valves.

TABLE 3.2.B (Cont)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
1	Core Spray Pump Start Timer	0.21<t<1 sec	Initiates sequential starting of CSCS pumps on any auto start.
1	LPCI Pump Start Timer	4.16<t<5.84 sec.	
1	LPCI Pump Start Timer	9.5<t<11.5 sec.	
1	Auto Blowdown Timer	≥94.4,≤115.6 sec.	In conjunction with Low Low Reactor Water Level, High Drywell Pressure and LPCI or Core Spray Pump running interlock, initiates Auto Blowdown.
2	ADS Drywell Pressure Bypass Timer	9≤t≤15.4 min.	Permits starting CS and LPCI pumps and actuating ADS SRV's if RPV water level is low and drywell pressure is not high.
2	RHR (LPCI) Pump Discharge Pressure Interlock	150 ± 10 psig	Defers ADS actuation pending confimation of Low Pressure Core Cooling System operation. (LPCI or Core Spray Pump running interlock.)
2	Core Spray Pump Discharge Pressure Interlock	150 ± 10 psig	
2	Emergency Bus Voltage Relay	20-25% of rated voltage resets at less than or equal to 50%	1. Permits closure of the Diesel Generator to an unloaded emergency bus. 2. Permits starting of CSCS 4 kV motors.

TABLE 3.2.B (Cont)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Startup Transformer Loss of Voltage	At 0 Volts between $0.96 \leq t \leq 1.34$ seconds Time Delay	1. Trips Startup Transformer to Emergency Bus Breaker. 2. Locks out automatic closure of Startup Transformer to Emergency Bus. 3. Initiates starting of Diesel Generators in conjunction with loss of auxiliary transformer. 4. Prevents simultaneous starting of CSCS components. 5. Starts load shedding logic for Diesel Operation in conjunction with (a) Low Low Reactor Water Level and Low Reactor Pressure or (b) High Drywell Pressure or (c) Core Standby Cooling System components in service in conjunction with Auxiliary Transformer breaker open.

TABLE 3.2.B (Cont)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Startup Transformer Degraded Voltage	3878.7V ± .51% with 10.24 ± 0.36 seconds time delay.	1. Trips Startup Transformer to Emergency Bus Breaker. 2. Locks out automatic closure of Startup Transformer to Emergency Bus. 3. Initiates starting of Diesel Generators in conjunction with loss of Auxiliary Transformer. 4. Prevents simultaneous starting of CSCS components. 5. Starts load shedding logic for Diesel Operation in conjunction with a) Low Low Reactor Water Level and Low Reactor Pressure or b) High Drywell Pressure or c) Core Standby Cooling System components in service in conjunction with Auxiliary Transformer breaker open.

TABLE 3.2.B (Cont)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
1	RHR (LPCI) Trip System bus power monitor	NA	Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	NA	Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	NA	Monitors availability of power to logic systems and valves.
1	HPCI Trip System bus power monitor	NA	Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	NA	Monitors availability of power to logic systems.
2	Recirculation Pump A d/p	≤2 psid	Operates RHR (LPCI) break detection logic which directs cooling water into unbroken recirculation loop.
2	Recirculation Pump B d/p	≤2 psid	
2	Recirculation Jet Pump Riser d/p A>B	0.5<p<1.5 psid	
1	Core Spray Sparger to Reactor Pressure Vessel d/p	-1(±1.5) psid	Alarm to detect Core Spray sparger pipe break.

TABLE 3.2.B (Cont)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Condensate Storage Tank Low Level	≥18" above tank zero	Provides interlock to HPCI pump suction valves.
2	Suppression Chamber High Level	≤1'11" below torus zero	
1	RCIC Turbine Steam Line High Flow	≤300% of rated steam flow (2)	
2	RCIC Turbine Compartment Wall	≤170°F (2)	
2	Torus Cavity Exhaust Duct	≤150°F (2)	
2	RCIC Valve Station Area Wall	≤200°F (2)	
4 (5)	RCIC Steam Line Lo-Press	100>P>50 psig (2)	
1	HPCI Turbine Steam Line High Flow	≤300% of rated flow (3)	
2	HPCI Turbine Compartment Exhaust Ducts	≤170°F (3)	
2	Torus Cavity Exhaust Duct	190 - 200°F (3)	
2	HPCI/RHR Valve Station Area Exhaust Duct	≤170°F (3)	

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS subsystem is required by Section 3.5 to be operable, there shall be two (Note 5) operable trip systems. If the first column cannot be met for one of the trip systems, that system shall be repaired or the reactor shall be placed in the Cold Shutdown Condition within 24 hours after this trip system is made or found to be inoperable.

2. Close isolation valves in RCIC subsystem.

3. Close isolation valves in HPCI subsystem.

4. Instrument set point corresponds to 79.96 inches above top of active fuel.

5. RCIC has only one trip system for these sensors.

TABLE 3.2.B.1

INSTRUMENTATION THAT MONITORS EMERGENCY BUS VOLTAGE

Minimum # of Operable Instrument Channels Per Trip system	Function	Setting	Remarks
1	Emergency 4160V Buses A5 & A6 Degraded Voltage Annunciation (1)	3958.5V + 0.5%-0.24% with 10.24 ± 0.36 seconds time delay	Alerts Operator to possible degraded voltage conditions. Provides permissive to initiate load shedding in conjunction with LOCA signal.

(1) In the event that the alarm system is determined inoperable, commence logging safety related bus voltage every ½ hour until such time as the alarm is restored to operable status.

TABLE 3.2.C.1

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Trip Function	Operable Channels per Trip Function Minimum	Available	Required Operational Conditions	Notes
APRM Upscale (Flow Biased)	4	6	Run	(1)
APRM Upscale	4	6	Startup/Refuel	(1)
APRM Inoperative	4	6	Run/Startup/Refuel	(1)
APRM Downscale	4	6	Run	(1)
Rod Block Monitor (Power Dependent)	2	2	Run, with limiting control rod pattern, and reactor power > LPSP	(2) (5)
Rod Block Monitor Inoperative	2	2	Run, with limiting control rod pattern, and reactor power > LPSP	(2) (5)
Rod Block Monitor Downscale	2	2	Run, with limiting control rod pattern, and reactor power > LPSP	(2) (5)
IRM Downscale	6	8	Startup/Refuel, except trip is bypassed when IRM is on its lowest range	(1)
IRM Detector not in Startup Position	6	8	Startup/Refuel, trip is bypassed when mode switch is placed in run	(1)
IRM Upscale	6	8	Startup/Refuel	(1)
IRM Inoperative	6	8	Startup/Refuel	(1)

TABLE 3.2.C.1 (Cont)

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Trip Function	Operable Instrument Channels per Trip Function Minimum	Available	Required Operational Conditions	Notes
SRM Detector not in Startup Position	3	4	Startup/Refuel, except trip is bypassed when SRM count rate is ≥ 100 counts/second or IRMs on Range 3 or above	(1) (4)
SRM Downscale	3	4	Startup/Refuel, except trip is bypassed when IRMs on Range 3 or above	(1) (4)
SRM Upscale	3	4	Startup/Refuel, except trip is bypassed when the IRM range switches are on Range 8 or above	(1) (4)
SRM Inoperative	3	4	Startup/Refuel, except trip is bypassed when the IRM range switches are on Range 8 or above	(1) (4)
Scram Discharge Instrument Volume Water Level - High	2	2	Run/Startup/Refuel	(3)
Scram Discharge Instrument Volume-Scram Trip Bypassed	1	1	Run/Startup/Refuel	(3)

TABLE 3.2.C.1 (Cont)

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Trip Function	Operable Instrument Channels per Trip Function Minimum	Available	Required Operational Conditions	Notes
Recirculation Flow Converter - Upscale	2	2	Run	(1)
Recirculation Flow Converter - Inoperative	2	2	Run	(1)
Recirculation Flow Converter - Comparator Mismatch	2	2	Run	(1)

NOTES FOR TABLE 3.2.C 1

1. With the number of operable channels:

   a. One less than required by the minimum operable channels per trip function requirement, restore an inoperable channel to operable status within 7 days or place an inoperable channel in the tripped condition within the next hour.

   b. Two or more less than required by the minimum operable channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. a. With one RBM Channel inoperable:

      (1) restore the inoperable RBM channel to operable status within 24 hours; otherwise place one rod block monitor channel in the tripped condition within the next hour, and;

      (2) prior to control rod withdrawal, perform an instrument function test of the operable RBM channel.

   b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

3. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

4. SRM operability requirements during core alterations are given in Technical Specification 3.10.

5. RBM operability is required in the run mode in the presence of a limiting rod pattern with reactor power greater than the RBM low power setpoint (LPSP). A limiting rod pattern exists when:

   MCPR < 1.40 for reactor power ≥ 90%

   MCPR < 1.70 for reactor power < 90%

   The allowable value for the LPSP is ≤ 29% of rated core thermal power.

TABLE 3.2.C-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

Trip Function	Trip Setpoint
APRM Upscale	(1) (2)
APRM Inoperative	Not Applicable
APRM Downscale	≥ 2.5 Indicated on Scale
Rod Block Monitor (Power Dependent)	(1) (3)
Rod Block Monitor Inoperative	Not Applicable
Rod Block Monitor Downscale	(1) (3)
IRM Downscale	≥ 5/125 of Full Scale
IRM Detector not in Startup Position	Not Applicable
IRM Upscale	≤ 108/125 of Full Scale
IRM Inoperative	Not Applicable
SRM Detector not in Startup Position	Not Applicable
SRM Downscale	≥ 3 counts/second
SRM Upscale	≤ 10^5 counts/second
SRM Inoperative	Not Applicable
Scram Discharge Instrument Volume Water Level - High	≤ 17 gallons
Scram Discharge Instrument Volume - Scram Trip Bypassed	Not Applicable
Recirculation Flow Converter - Upscale	≤ 120/125 of Full Scale
Recirculation Flow Converter - Inoperative	Not Applicable
Recirculation Flow Converter - Comparator Mismatch	≤ 8% Flow Deviation

(1) The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

(2) When the reactor mode switch is in the refuel or startup positions, the APRM rod block trip setpoint shall be less than or equal to 13% of rated thermal power, but always less than the APRM flux scram trip setting.

(3) The RBM bypass time delay (t_{d2}) shall be < 2.0 seconds.

TABLE 3.2.D

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE

Minimum # of Operable Instrument Channels Per Trip system (1)	Trip Function	Trip Level Setting	Action (2)
2	Refuel Area Exhaust Monitors	Upscale, <100 mr/hr	A or B
2	Refuel Area Exhaust Monitors	Downscale	A or B

NOTES FOR TABLE 3.2.D

1. Whenever the systems are required to be operable, there shall be two operable or tripped trip systems. If this cannot be met, the indicated action shall be taken.

1. Action

   A. Cease operation of the refueling equipment.
   B. Isolate secondary containment and start the standby gas treatment system.

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TABLE 3.2.F

SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Parameter	Type Indication and Range	Notes
2	640-29A & B	Reactor Water Level	Indicator 0-60"	(1) (2) (3)
2	640-25A & B	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	TRU-9044 TRU-9045	Dyrwell Pressure	Recorder 0-80 psia	(1) (2) (3)
2	TRU-9044 TI-9019	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
2	TRU-9045 TI-9018	Suppression Chamber Air Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
2	LR-5038 LR-5049	Suppression Chamber Water Level	Recorder 0-32"	(1) (2) (3)
1	NA	Control Rod Position	28 Volt Indicating Lights	(1) (2) (3) (4)
1	NA	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power	(1) (2) (3) (4)

TABLE 3.2.F (Cont)

SURVEILLANCE INSTRUMENTATION

Minimum # of Channels	Operable Instrument Instrument #	Parameter	Type Indication and Range	Notes
2	(TI-5021-01A (TRU-5021-01A (((TI-5022-01B (TRU-5022-01B	Suppression Chamber Water Temperature Suppression Chamber Water Temperature	Dual Indicator/ Multipoint Recorder 30-230°F (Bulk/Local) Dual Indicator/ Multipoint Recorder 30-230°F (Bulk/Local)	(4) (1) (2) (3) (4) (1) (2) (3)
1	PI-5021	Drywell/Torus Diff. Pressure	Indicator -.25 - +3.0 psig	(1) (2) (3) (4)
1	(PI-5067A (PI-5067B	Drywell Pressure Torus Pressure	Indicator -.25 - +3.0 psig Indicator -1.0 - +2.0 psig	 (1) (2) (3) (4)
1/Valve	(a)Primary (or (5) (b)Backup	Safety/Relief Valve Position	a) Acoustic monitor b) Thermocouple	(5)
1/Valve	(a)Primary (or (5) (b)Backup	Safety Valve Position Indicator	a) Acoustic monitor b) Thermocouple	(5)
1/Valve	See Note (6)	Tail Pipe Temperature Indication	Thermocouple	(6)
2	(LI 1001-604A (LR 1001-604A ((LI 1001-604B (LR 1001-604B	Torus Water Level (Wide Range) Torus Water Level (Wide Range)	Indicator/Multipoint Recorder 0-300" H_2O Indicator/Multipoint Recorder 0-300"H_2O	(4) (1) (2) (3) (4) (1) (2) (3)

TABLE 3.2.F (Cont)

SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Parameter	Type Indication and Range	Notes
2	(PI 1001-600A (PR 1001-600A ((PI 1001-600B (PR 1001-600B	Containment Pressure, (High Range) Containment Pressure, (High Range)	Indicator/Multipoint Recorder 0-225 psig Indicator/Multipoint Recorder 0-225 psig	(4) (1) (2) (3) (4) (1) (2) (3)
2	(PI 1001-601A (PR 1001-600A ((PI 1001-601B (PR 1001-600B	Containment Pressure, (Low Range) Containment Pressure, (Low Range)	Indicator/Multipoint Recorder -5 to 5 psig Indicator/Multipoint Recorder -5 to 5 psig	(4) (1) (2) (3) (4) (1) (2) (3)
2	(RIT 1001-606A (RIT 100'-606B (RR 1001-606A (RR 1001-606B	Containment High Radiation (Drywell)	Monitor/Multipoint Recorder 1 to $1x10^7$ R/hr	(4) (7)
1	RI 1001-609 RR 1001-608	Reactor Building Vent	Indicator/Multipoint Recorder 10^{-1} to 10^4 R/hr	(4) (7)
1	RI 1001-608 RR 1001-608	Main Stack Vent	Indicator/Multipoint Recorder 10^{-1} to 10^4 R/hr	(4) (7)
1	RI 1001-610 RR 1001-608	Turbine Building Vent	Indicator/Multipoint Recorder 10^{-1} to 10^4 R/hr	(4) (7)

NOTES FOR TABLE 3.2.F

(1) With less than the minimum number of instrument channels, restore the inoperable channel(s) within 30 days.

(2) With the instrument channel(s) providing no indication to the control room, restore the indication to the control room within seven days.

(3) If the requirements of notes (1) or (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) At a minimum, the primary or back-up* parameter indicators shall be operable for each valve when the valves are required to be operable. With both primary and backup* instrument channels inoperable either return one (1) channel to operable status within 31 days or be in a shutdown mode within 24 hours.

The following instruments are associated with the safety/relief and safety valves:

Valve	Primary Acoustic Monitor	Secondary Tail Pipe Temperature Thermocouple
203-3A	ZT-203-3A	TE6271 *
203-3B	ZT-203-3B	TE6272 *
203-3C	ZT-203-3C	TE6273 *
203-3D	ZT-203-3D	TE6276 *
203-4A	ZT-203-4A	TE6274-B
203-4B	ZT-203-4B	TE6275-B

* See Note (6)

(6) At a minimum, for thermocouples providing SRV tail pipe temperature, one of the dual thermocouples will be operable for each SRV when the valves are required to be operable. If a thermocouple becomes inoperable, it shall be returned to an operable condition within 31 days or the reactor shall be placed in a shutdown mode within 24 hours.

(7) With less than the minimum number of operable instrument channels, restore the inoperable channels to operable status within 7 days or prepare and submit a special report to the Commission within 14 days of the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the channels to operable status.

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP
AND
ALTERNATE ROD INSERTION

Minimum Number of Operable or Tripped Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting
2	High Reactor Dome Pressure	1175 ± 5 psig
2	Low-Low Reactor Water Level	≥ -46.3" indicated level

Actions (1) There shall be two (2) operable trip systems for each function.

(a) If the minimum number of operable or tripped instrument channels for one (1) trip system cannot be met, restore the trip system to operable status within 14 days or be in at least hot shutdown within 24 hours.

(b) If the minimum operability conditions (1.a) cannot be met for both (2) trip systems, be in at least hot shutdown within 24 hours.

TABLE 3.2.H

DRYWELL TEMPERATURE SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Elements	Instrument #	Nominal Instrument Elevation	Type	Note
Above 40 Feet Elevation				
1/ELEV	TE-5050A 1/2 TE-5050B 1/2	80' 80'	RTD	(1) (2) (4)
1/ELEV	TE-5050C 1/2 TE-5050D 1/2	87' 87'	RTD	(1) (2) (4)
1/ELEV	TE-5050E 1/2 TE-5050F 1/2	60' 60'	RTD	(1) (2) (4)
Below 40 Feet Elevation				
1/ELEV	TE-5050G 1/2 TE-5050H 1/2	41'	RTD	(1) (3) (4)
1/ELEV	TE-5050J 1/2 OR TE-5050K 1/2	32'	RTD	(1) (3) (4)

Notes:

1. The 5050 series temperature elements are dual-elements.

2. At least one element of one RTD on each elevation shall be operable.

3. At least one element of one RTD on elevation 41 and one element of one of the RTD's at nominal elevation 32 shall be operable.

4. If the minimum number of operable RTD's as specified in Note 2 and 3 above are not available and cannot be made available within 24 hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours of shutdown initiation.

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TABLE 4.2.A

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

	Instrument Channel (5)	Instrument Functional Test		Calibration Frequency	Instrument Check
1)	Reactor High Pressure	(1)		Once/3 months	None
2)	Reactor Low-Low Water Level	Once/3 months	(7)	(7)	Once/day
3)	Reactor High Water Level	Once/3 months	(7)	(7)	Once/day
4)	Main Steam High Temp.	(1)		Once/3 months	None
5)	Main Steam High Flow	Once/3 months	(7)	(7)	Once/day
6)	Main Steam Low Pressure	Once/3 months	(7)	(7)	Once/day
7)	Reactor Water Cleanup High Flow	(1)		Once/3 months	Once/day
8)	Reactor Water Cleanup High Temp	(1)		Once/3 months	None

Logic System Functional Test (4) (6)		Frequency
1)	Main Steam Line Isolation Vvs. Main Steam Line Drain Vvs. Reactor Water Sample Vvs.	Once/Operating Cycle
2)	RHR - Isolation Vv. Control Shutdown Cooling Vvs. Head Spray Discharge to Radwaste	Once/Operating Cycle
3)	Reactor Water Cleanup Isolation	Once/Operating Cycle
4)	Drywell Isolation Vvs. TIP Withdrawal Atmospheric Control Vvs. Sump Drain Valves	Once/Operating Cycle
5)	Standby Gas Treatment System Reactor Building Isolation	Once/Operating Cycle

TABLE 4.2.B

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

	Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1)	Reactor Water Level	(1) (7)	(7)	Once/day
2)	Drywell Pressure	(1) (7)	(7)	Once/day
3)	Reactor Pressure	(1) (7)	(7)	Once/day
4)	Auto Sequencing Timers	NA	Once/Operating Cycle	None
5)	ADS - LPCI or CS Pump Disch. Pressure Interlock	(1)	Once/3 months	None
6)	Start-up Transf. (4160V)			
	a. Loss of Voltage Relays	Monthly	Once/Operating Cycle	None
	b. Degraded Voltage Relays	Monthly	Once/Operating Cycle	None
7)	Trip System Bus Power Monitors	Once/Operating Cycle	NA	Once/day
8)	Recirculation System d/p	(1)	Once/3 months	Once/day
9)	Core Spray Sparger d/p	NA	Once/18 months	Once/day
10)	Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
11)	Steam Line High Temp. (HPCI & RCIC)	(1)	Once/3 months	None
12)	Safeguards Area High Temp.	(1)	Once/3 months	None

TABLE 4.2.B (Cont)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

	Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
13)	RCIC Steam Line Low Pressure	(1)	Once/3 months	None
14)	HPCI Suction Tank Levels	(1)	Once/3 months	None
15)	Emergency 4160V Buses A5 & A6 Loss of Voltage Relays	Monthly	Once/Operating Cycle	None

TABLE 4.2.B (Cont)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Logical System Functional Test (4) (6)	Frequency	Remarks
1) Core Spray Subsystem	Once/Operating Cycle	
2) Low Press. Coolant Injection Subsystem	Once/Operating Cycle	
3) Containment Spray Subsystem	Once/Operating Cycle	
4) HPCI Subsystem	Once/Operating Cycle	
5) HPCI Subsystem Auto Isolation	Once/Operating Cycle	
6) ADS Subsystem	Once/Operating Cycle	
7) RCIC Subsystem Auto Isolation	Once/Operating Cycle	
8) Diesel Generator Initiation	Once/Operating Cycle	
9) Area Cooling for Safeguard System	Once/Operating Cycle	

TABLE 4.2.C

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
APRM - Downscale	Once/3 Months	Once/3 Months	Once/Day
APRM - Upscale	Once/3 Months	Once/3 Months	Once/Day
APRM - Inoperative	Once/3 Months	Not Applicable	Once/Day
IRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
IRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
IRM - Inoperative	(2) (3)	Not Applicable	(2)
RBM - Upscale	Once/3 Months	Once/6 Months	Once/Day
RBM - Downscale	Once/3 Months	Once/6 Months	Once/Day
RBM - Inoperative	Once/3 Months	Not Applicable	Once/Day
SRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
SRM - Inoperative	(2) (3)	Not Applicable	(2)
SRM - Detector Not in Startup Position	(2) (3)	Not Applicable	(2)
SRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
IRM - Detector Not in Startup Position	(2) (3)	Not Applicable	(2)
Scram Discharge Instrument Volume Water Level-High	Once/3 Months	Refuel	Not Applicable
Scram Discharge Instrument Volume-Scram Trip Bypassed	Once/3 Months	Not Applicable	Not Applicable
Recirculation Flow Converter	Not Applicable	Once/Operating Cycle	Once/Day
Recirculation Flow Converter-Upscale	Once/3 Months	Once/3 Months	Once/Day
Recirculation Flow Converter-Inoperative	Once/3 Months	Not Applicable	Once/Day
Recirculation Flow Converter-Comparator Off Limits	Once/3 Months	Once/3 Months	Once/Day
Recirculation Flow Process Instruments	Not Applicable	Once/Operating Cycle	Once/Day

Logic System Functional Test (4) (6)

System Logic Check	Once/Operating Cycle

TABLE 4.2.D

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS

Instrument Channels	Instrument Functional Test	Calibration	Instrument Check (2)
1) Refuel Area Exhaust Monitors - Upscale	(1)	Once/3 months	Once/day
2) Refuel Area Exhaust Monitors - Downscale	(1)	Once/3 months	Once/day

Logic System Functional Test (4) (6)	Frequency
1) Reactor Building Isolation	Once/Operating Cycle
2) Standby Gas Treatment System Actuation	Once/Operating Cycle

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

	Instrument Channel	Calibration Frequency	Instrument Check
1)	Reactor Water Level	Each Refueling Outage	Each Shift
2)	Reactor Pressure	Each Refueling Outage	Each Shift
3)	Drywell Pressure	Each Refueling Outage	Each Shift
4)	Drywell Temperature	Once/6 Months	Each Shift
5)	Suppression Chamber Temperature	Once/6 Months	Each Shift
6)	Suppression Chamber Water Level	Once/6 Months	Each Shift
7)	Control Rod Position	NA	Each Shift
8)	Neutron Monitoring	(2)	Each Shift
9)	Drywell/Torus Differential Pressure	Once/6 Months	Each Shift
10)	Drywell Pressure Torus Pressure	Once/6 Months Once/6 Months	Each Shift
11)	Safety/Relief Valve Position Indicator (Primary/Secondary)	Each refueling outage	Once each day
12)	Safety Valve Position Indicator (Primary/Secondary)	Each refueling outage	Once each day

TABLE 4.2.F (Cont)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

	Instrument Channel	Calibration Frequency	Instrument Check
13)	Torus Water Level (Wide Range)	Each refueling outage	Once every 30 days
14)	Containment Pressure	Each refueling outage	Once every 30 days
15)	Containment High Radiation	Each refueling outage	Once every 30 days
16)	Reactor Building Vent Radiation Monitor	Each refueling outage	Once every 30 days
17)	Main Stack Vent Radiation Monitor	Each refueling outage	Once every 30 days
18)	Turbine Building Vent Radiation Monitor	Each refueling outage	Once every 30 days

PNPS
TABLE 4.2.G

MINIMUM TEST AND CALIBRATION FREQUENCY FOR
ATWS RPT/ARI INSTRUMENTATION

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
1. Reactor High Pressure	(1) (7)	(7)	Once/day
2. Reactor Low-Low Water Level	(1) (7)	(7)	Once/day

PNPS
TABLE 4.2.H

MINIMUM TEST & CALIBRATION FREQUENCY FOR DRYWELL TEMPERATURE SURVEILLANCE INSTRUMENTATION

Instrument Channels/ Nominal Elevation	Calibration Frequency	Instrument Check
80 Feet	Each Refueling Outage	Once per Shift
87 Feet	Each Refueling Outage	Once per Shift
60 Feet	Each Refueling Outage	Once per Shift
41 Feet	Each Refueling Outage	Once per Shift
32 Feet	Each Refueling Outage	Once per Shift

NOTES FOR TABLES 4.2.A THROUGH 4.2.G

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations of IRMs and SRMs shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level and high drywell pressure are not included on Table 4.2.A since they are tested on Tables 4.1.1 and 4.1.2.

6. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

7. Calibration of analog trip units will be performed concurrent with functional testing. The functional test will consist of injecting a simulated electrical signal into the measurement channel. Calibration of associated analog transmitters will be performed each refueling outage.

BASES:

3 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation closes all isolation valves except those in Groups 1, 4 and 5. This trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low low reactor water level instrumentation closes the Main Steam Line Isolation Valves, Main Steam Drain Valves, Recirc Sample Valves (Group 1) activates the CSCS subsystems, starts the emergency diesel generators and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that no fuel damage will occur and so that post accident cooling can be accomplished and the guidelines of 10CFR100 will not be violated. For large breaks

3.2 PROTECTIVE INSTRUMENTATION (Cont)

up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of Group 1 isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, steam flow trip setting in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain approximately 1000°F and release of radioactivity to the environs is well below 10CFR100 guidelines.

Temperature monitoring instrumentation is provided in the main steam line tunnel and the turbine basement to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 170°F for the main steam line tunnel detector is low enough to detect leaks on the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

Pressure instrumentation is provided to close the main steam isolation valves in the RUN mode before the reactor pressure drops below 785 psig. This function is primarily intended to prevent excessive vessel depressuirzation in the event of a malfunction of the nuclear system pressure regulator. This function also provides automatic protection of the low-pressure core-thermal-power safety limit (25% of rated core thermal power for reactor pressure < 785 psig). In the Refuel or Startup Mode, the inventory loss associated with such a malfunction would be limited by closure of the Main Steam Isolation Valves due to either high or low reactor water level; no fuel would be uncovered. This function is not required to satisfy any safety design bases.

3.2 PROTECTIVE INSTRUMENTATION (Cont)

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic, and all sensors are required to be operable.

Temperature is monitored at three (3) locations with four (4) temperature sensors at each location. Two (2) sensors at each location are powered by "A" direct current control bus and two (2) by "B" direct current control bus. Each pair of sensors, e.g., "A" or "B", at each location are physically separated and the tripping of either "A" or "B" bus sensor will actuate HPCI isolation valves.

The trip settings of ≤ 300% of design flow for high flow and 200°F or 170°F, depending on sensor location, for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of ≤ 300% for high flow and 200°F, 170°F and 150°F, depending on sensor location, for temperature are based on the same criteria as the HPCI.

The Reactor Water Cleanup System high flow and temperature instrumentation are arranged similar as that for the HPCI. The trip settings are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, two RMB's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for not longer than 24 hours without significantly increasing the risk of an inadvertent control rod withdrawal.

Reactor power may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point, thereby possibly avoiding an APRM Scram. The rod block setpoint is automatically reduced with recirculation flow to form the upper boundary of the PNPS power/flow map. The flow biased APRM rod block is not necessary to prohibit fuel damage and is not included in the analysis of anticipated transients.

3.2 PROTECTIVE INSTRUMENTATION (Cont)

The RBM rod block function provides local protection of the core, for a single rod withdrawal error from a limiting control rod pattern.

The RBM bypass time delay (t_{d2}) is the delay between the time the signal is normalized to the reference signal and the time the signal is passed to the trip logic. Control rod withdrawal is unrestricted during this interval. The RBM bypass time delay is low enough to assure that control rod movement is minimized during the time RBM trips are bypassed.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM, RBM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are as shown in Table 3.2.C-2.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Four radiation monitors are provided which initiate the Reactor Building Isolation and Control System and operation of the standby gas treatment system. The instrument channels monitor the radiation from the refueling area ventilation exhaust ducts.

Four instrument channels are arranged in a 1 out of 2 twice trip logic.

3.2 PROTECTIVE INSTRUMENTATION (Cont)

Trip settings of < 100 mr/hr for the monitors in the refueling area ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

For most parameters monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between these two (2) channels, a near continuous surveillance of instrument performance is available. Meaningful deviation in comparative readings of these instruments will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The Safety - Safety/Relief Valve position indication instrumentation provides the operator with information on selected plant parameters to monitor and assess these variables during and following an accident.

In response to NUREG-0737, modifications were made to the ADS logic to extend automatic ADS operation to a class of transients that involve slowly uncovering the core without depressurizing the vessel or pressurizing the drywell. These transients were analyzed assuming no high pressure injection systems (feedwater, HPCI or RCIC) are available. Only ADS is considered available to depressurize the vessel, permitting operation of LPCI. The transients generally involve pipe breaks outside containment. Automatic ADS would not occur on low water level because high drywell pressure would not be present and ADS logic has a high drywell pressure permissive. The modification added a timer to the ADS logic which bypasses the high drywell pressure permissive, and a manual inhibit switch which allows the operator to inhibit automatic ADS initiation for events where automatic initiation is not desirable.

An analysis was performed to determine an upper time limit on the bypass timer. The goal was to ensure ADS is automatically initiated in time to prevent peak clad temperature (PCT) from exceeding 1500°F for a limiting break, which was determined to be a Reactor Water Cleanup line break. The analysis concluded that there are 18 minutes between the low water level initiation of the timer and the heatup of the cladding to the limit. Since the logic includes a 2 minute delay already, the bypass timer upper limit can not be more than 16 minutes, which provides a conservative margin for PCT and allows sufficient time for operator intervention if required. A minimum time delay is incorporated to allow RPV water level to recover, resetting the timer and preventing depressurization. The choice of a timer setting of 11 minutes places the setting in the middle and provides maximum tolerance from either limit. (Reference: GE Report "Bypass Timer Calculation for the ADS/ECCS Modification for Pilgrim Station" December 16, 1986).

3.2 PROTECTIVE INSTRUMENTATION (Cont)

The recirculation pump trip/alternate rod insertion systems are consistent with the "Monticello RPT/ARI" design described in NEDO-25016 (Reference 1) as referenced by the NRC as an acceptable design (Reference 2) for RPT. Reference 1 provides both system descriptions and performance analyses. The pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction increases core voiding providing a negative reactivity feedback. High pressure sensors and low water level sensors initiate the trip. The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated in this unlikely postulated event. Requiring the trip to be operable only when in the RUN mode is therefore conservative. The low water level trip function includes a time delay of nine (9) seconds ± one (1) second to avoid increasing the consequences of a postulated LOCA. This delay has an insignificant effect on ATWS consequences.

Alternate rod insertion utlizes the same initiation logic and functions as RPT and provides a diverse means of initiating a reactor scram. ARI uses sensors diverse from the reactor protection system to depressurize the scram pilot air header, which in turn causes all control rods to be inserted.

References

1. NEDO-25016, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant," September 1976.

2. NUREG-0460, Volume 3, December 1978.

Drywell Temperature

The drywell temperature limitations of Specification 3.2.H.1 ensure that safety related equipment will not be subjected to excess temperature. Exposure to excessive temperatures may degrade equipment and can cause loss of its operability.

The temperature elements for monitoring drywell temperature specified in Table 3.2.H were chosen on the basis of their reliability, location, and their redundancy (dual - element RTD's). These temperature elements are the primary elements used for the PCILRT.

The "nominal instrument elevations" provided in Tables 3.2.H and 4.2.H assist personnel in locating the instruments for surveillance and maintenance purposes and define the approximate containment region to be monitored. The "nominal instrument elevations" are not intended to provide a precise instrument location.

3.2 PROTECTIVE INSTRUMENTATION (Cont)

The temperature limits specified in 3.2.H.1 are based on the BECo report entitled Drywell Temperature Report, dated January 28, 1982. The limits derived from this report take into consideration the long-term effects of ambient temperature on equipment design limits and material degradation of components required for accident mitigation or plant shutdown. The evaluation process addressed the actual assessment of potential damage and the determination of equipment status from the standpoint of both qualification integrity (for safety-related equipment) and reliability to perform its intended function.

If the drywell temperature exceeds the limits specified in 3.2.H.1 an engineering evaluation must be initiated in order to determine whether any safety related component has been adversely affected.

The limiting drywell temperature value of 215°F (Section 3.2.H.2) was selected as to guarantee that ECCS trips occur on/or before present Technical Specification values.

The time interval of 30 minutes between successive drywell temperature instrument readings (Section 3.2.H.1) was selected so as to guarantee that ECCS trips occur on/before present Technical Specification values in the event of a drywell temperature excursion in excess of 215°F.

The instrument check interval of once per shift provides adequate assurance of equipment operability based upon engineering judgement.

The instrumentation listed in Table 4.2.A thru 4.2.H will be functionally tested and/or calibrated at regularly scheduled intervals. The same design reliability goal as the Reactor Protection System of 0.99999 is generally applied for all applications of (1 out of 2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bi-stable trips associated with analog sensors and amplifiers are tested once/week.

BASES:

4.2 PROTECTIVE INSTRUMENTATION

Conservatively assuming that those instruments which have their contacts arranged in 1 out of n logic cannot be used during a testing sequence, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = \sqrt{\frac{2t}{r}}$$

Where: i = the optimum interval between tests.

t = the time the trip contacts are disabled from performing their function while the test is in progress.

r = the expected failure rate of the relays.

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hour. Using this data and the above operation, the optimum test interval is

$$i = \sqrt{\frac{2(0.5)}{10^{-6}}} = 1 \times 10^{3} \text{ hours}$$

$$= 40 \text{ days}$$

For additional margin a test interval of once per month will be used initially.

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

(7) UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, page 10, Equation (24), Lawrence Radiation Laboratory.

4.2 PROTECTIVE INSTRUMENTATION (Cont)

The above calculated test interval optimizes each individual channel, considering it to be independent of all others. As an example, assume that there are two channels with an individual technician assigned to each. Each technician tests his channel at the optimum frequency, but the two technicians are not allowed to communicate so that one can advise the other that his channel is under test. Under these conditions, it is possible for both channels to be under test simultaneously. Now, assume that the technicians are required to communicate and that two channels are never tested at the same time.

Forbidding simultaneous testing improves the availability of the system over that which would be achieved by testing each channel independently. These one out of n trip systems will be tested one at a time in order to take advantage of this inherent improvement in availability.

Optimizing each channel independently may not truly optimize the system considering the overall rules of system operation. However, true system optimization is a complex problem. The optimums are broad, not sharp, and optimizing the individual channels is generally adequate for the system.

The formula given above minimizes the unavailability of a single channel which must be bypassed during testing. The minimization of the availability is illustrated by Curve No. 1 of Figure 4.2.2 which assumes that a channel has a failure rate of 0.1×10^{-6}/hour and that 0.5 hours is required to test it. The unavailability is a minimum at a test interval i, of 3.16×10^{3} hours.

If two similar channels are used in a 1 out of 2 configuration, the test interval for minimum unavailability changes as a function of the rules for testing. The simplest case is to test each one independent of the other. In this case, there is assumed to be a finite probability that both may be bypassed at one time. This case is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

A more unusual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

4.2 PROTECTIVE INSTRUMENTATION (Cont)

The most likely case would be to stipulate that one channel be bypassed, tested, and restored, and then immediately following, the second channel be bypassed, tested and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the refueling area ventilation duct which initiate building isolation and Standby Gas Treatment operation are arranged in two 1 out of 2 logic systems. The bases given above apply here also and were used to arrive at the functional testing frequency. Based on experience with instruments of similar design, a testing interval of once every three months has been found adequate.

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

The instrumentation which is required for the recirculation pump trip and alternate rod insertion systems incorporate analog transmitters. The transmitter calibration frequency is once per refueling outage, which is consistent with both the equipment capabilities and the requirements for similar equipment used at Pilgrim. The Trip Unit Calibration and Instrument Functional Test is specified at monthly, which is the same frequency specified for other similar protective devices. An instrument check is specified at once per day; this is considered to be an appropriate frequency, commensurate with the design applications and the fact that the recirculation pump trip and alternate rod insertion systems are backups to existing protective instrumentation.

4.2 PROTECTIVE INSTRUMENTATION (Cont)

Control Rod Block and PCIS instrumentation common to RPS instrumentation have surveillance intervals and maintenance outage times selected in accordance with NEDC-30851P-A, Supplements 1 and 2 as approved by the NRC and documented in SERs (letters to D. N. Grace from C. E. Rossi dated September 22, 1988 and January 6, 1989).

A logic system functional test interval of 24 months was selected to minimize the frequency of safety system inoperability due to testing and to minimize the potential for inadvertent safety system trips and their attendant transients.

PNPS
FIGURE 4.2-1

GRAPHICAL AID IN THE SELECTION OF
AN ADEQUATE INTERVAL BETWEEN TESTS

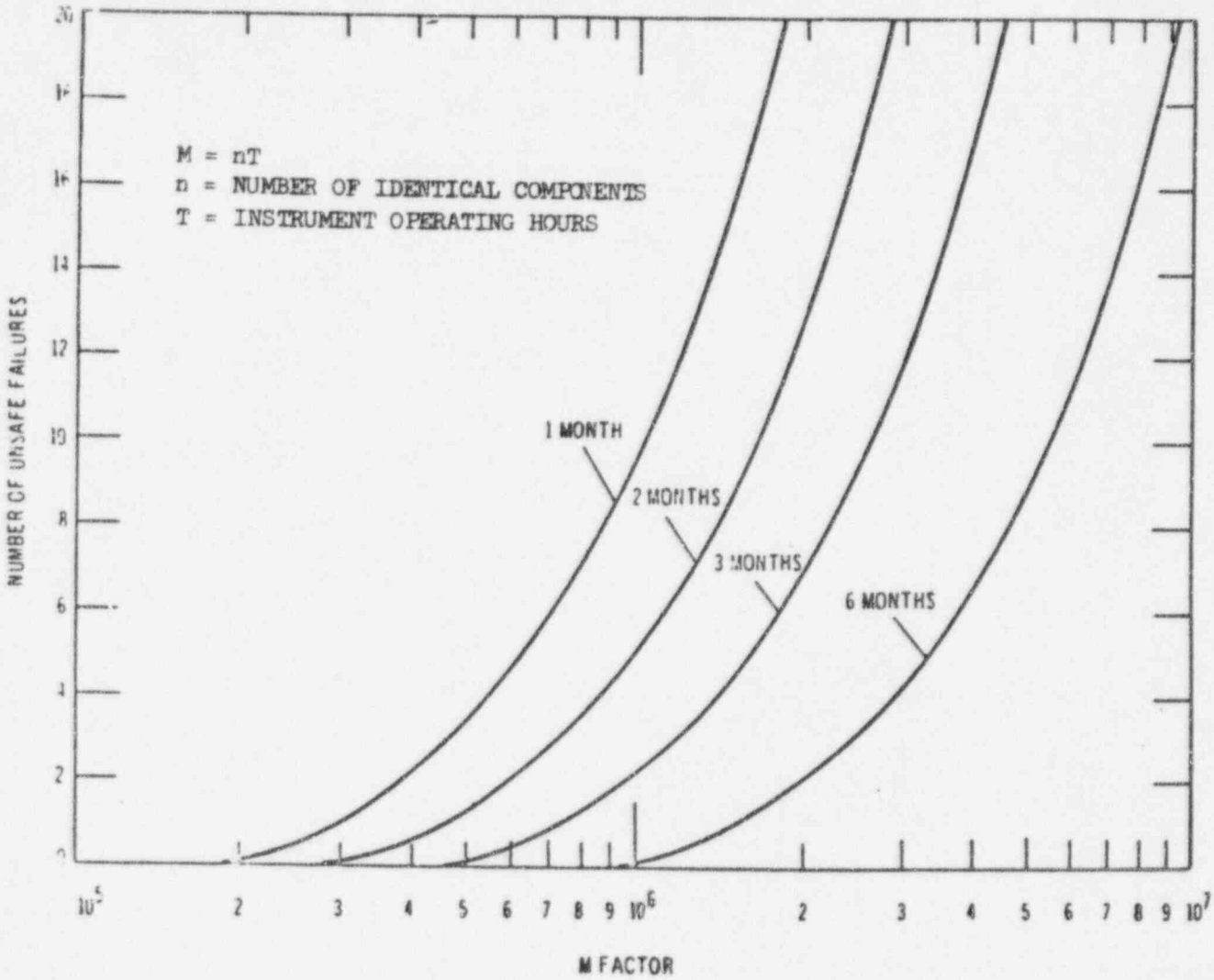
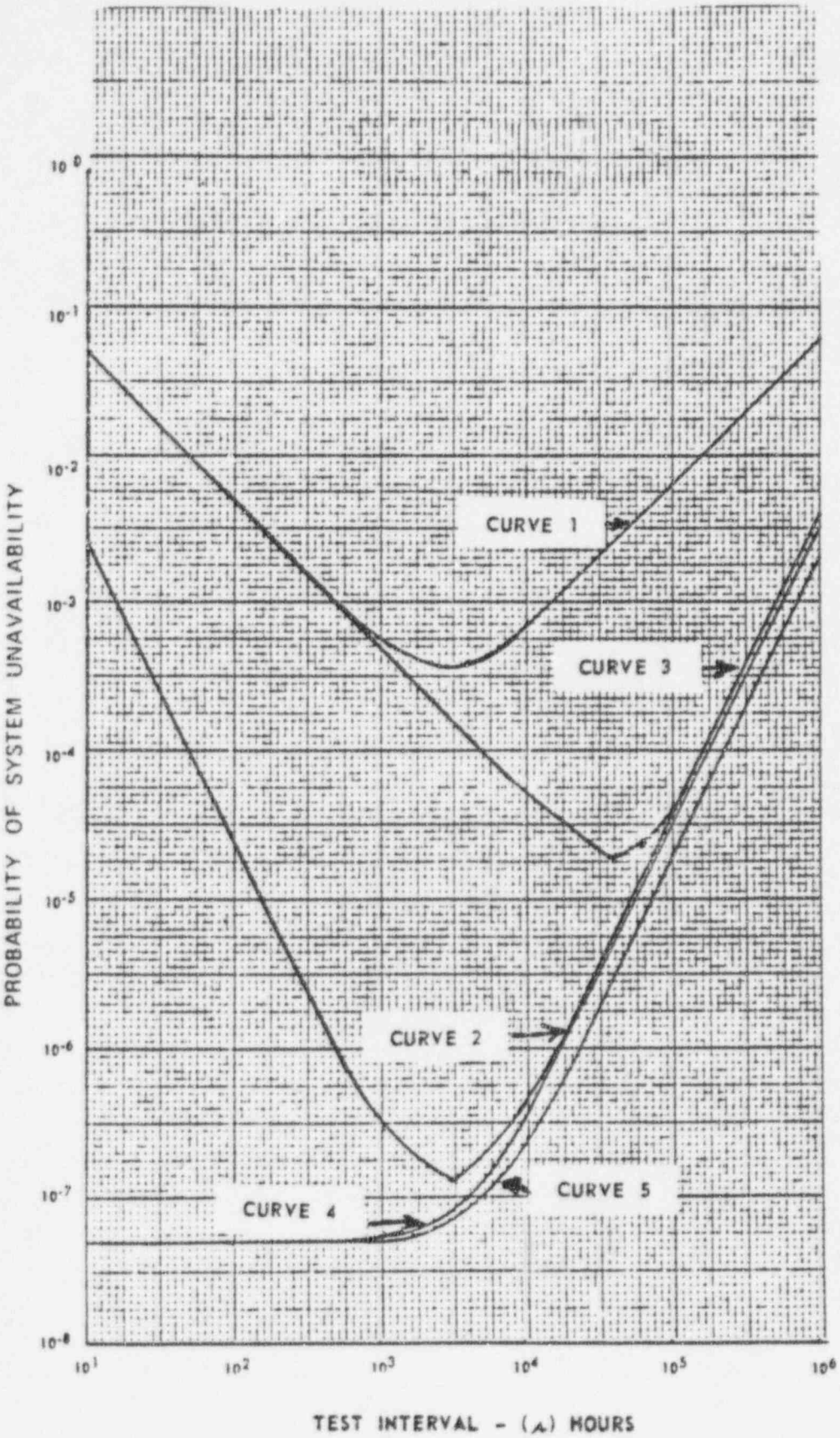


FIGURE 4.2-2

SYSTEM UNAVAILABILITY



LIMITING CONDITION FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operating cycle with the strongest operable control rod in its full-out position and all other operable rods fully inserted.

2. Reactivity margin - inoperable control rods

a. Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing.

SURVEILLANCE REQUIREMENT

4.3 REACTIVITY CONTROL

Applicability:

Applies to the surveillance requirements of the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25 percent Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully witdrawn and all other operable rods fully inserted.

2. Reactivity margin - inoperable control rods

Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with three or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance

3.3 REACTIVITY CONTROL (Cont)

A. Reactivity Limitations (Cont)

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically and the control rods shall be in such positions that Specification 3.3.A.1 is met.

c. Control rod drives which are fully inserted and electrically disarmed shall not be considered inoperable.

d. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be moved with control rod drive pressure they need not be disarmed electrically.

e. During reactor power operation, the number of inoperable control rods shall not exceed eight. Specification 3.3.A.1 must be met at all times.

f. If plant operation is continued with three (3) or more control rods with maximum scram insertion times in excess of 7.00 seconds, perform the surveillance requirement of Section 4.3.C.2 at least once per 60 days.

B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional or control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

4.3 REACTIVITY CONTROL (Cont)

A. Reactivity Limitations (Cont)

need not be completed within 24 hours if the number of inoperable rods has been reduced to less than three and if it has been demonstrated that control rod drive mechanism collet housing failure is not cause of an immovable control rod.

B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

a. When the rod is withdrawn the first time subsequent to each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is critical shall be performed to verify instrumentation response.

LIMITING CONDITION FOR OPERATION

3.3 REACTIVITY CONTROL (Cont)

B. Control Rods (Cont)

2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

3. a. No control rods shall be moved when the reactor is below 20% rated power, except to shutdown the reactor, unless the Rod Worth Minimizer (RWM) is operable. A maximum of two rods may be moved below 20% design power when the RWM is inoperable if all other rods except those which cannot be moved with control rod drive pressure are fully inserted.

b. Control rod patterns and the sequence of withdrawal or insertion shall be established such that:

1) when the reactor is critical and below 10% design power the maximum worth of any insequence control rod which is not electrically disarmed is less than 0.010 delta k.

2) and when the reactor is above 20% design power the maximum worth of any control rod, including allowance for a single operator error, is less than 0.020 delta k.

SURVEILLANCE REQUIREMENT

4.3 REACTIVITY CONTROL (Cont)

B. Control Rods (Cont)

b. When the rod is fully withdrawn the first time subsequent to each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup or insertion to reduce power below 20% of the operability of the Rod Worth Minimizer (RWM) shall be verified by:

a. verifying the correctness of the control rod withdrawal sequence input to the RWM computer.

b. performing the RWM computer diagnostic test.

c. verifying the annunciation of the selection errors of at least one out-of-sequence control rod in each distinct RWM group.

d. verifying the rod block function of an out-of-sequence control rod which is withdrawn no more than three notches.

LIMITING CONDITION FOR OPERATION

3.3 REACTIVITY CONTROL (Cont)

B. Control Rods (Cont)

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. The RBM shall be operable as required in Table 3.2.C-1, or control rod withdrawal shall be blocked.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
10	.55
30	1.275
50	2.00
90	3.50

SURVEILLANCE REQUIREMENT

4.3 REACTIVITY CONTROL (Cont)

B. Control Rods (Cont)

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

C. Scram Insertion Times

1. Following each refueling outage, or after a reactor shutdown that is greater than 120 days, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished with the nuclear system pressure above 950 psig, the measured scram insertion time shall be extrapolated to reactor pressures above 950 psig using previously determined correlations. Testing of all operable control rods shall be completed prior to exceeding 40% rated thermal power.

LIMITING CONDITION FOR OPERATION

3.3 REACTIVITY CONTROL (Cont)

C. Scram Insertion Time (Cont)

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Time Sec.
10	.58
30	1.35
50	2.12
90	3.71

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator.

2. Directional control valve electrically disarmed while in a non-fully inserted position.

3. Scram insertion time greater than the maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.

SURVEILLANCE REQUIREMENT

4.3 REACTIVITY CONTROL (Cont)

C. Scram Insertion Time (Cont)

2. Within each 120 days of operation, a minimum of 10% of the control rod drives, on a rotating basis, shall be scram tested as in 4.3.C.1. An evaluation shall be completed every 120 days of operation to provide reasonable assurance that proper performance is being maintained.

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

LIMITING CONDITION FOR OPERATION

3.3 REACTIVITY CONTROL (Cont)

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% ΔK. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken if such actions are appropriate.

F. If Specifications 3.3.A through D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours. Specifications 3.3.A through D above do not apply when there is no fuel in the reactor vessel.

G. Scram Discharge Volume

1. The scram discharge volume drain & vent valves shall be operable whenever more than one operable control rod is withdrawn.

2. If any of the scram discharge volume drain or vent valves are made or found inoperable an orderly shutdown shall be initiated and the reactor shall be in Cold Shutdown within 24 hours.

SURVEILLANCE REQUIREMENT

4.3 REACTIVITY CONTROL (Cont)

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

F. Not Used

G. Scram Discharge Volume

1. Scram discharge volume drain and vent valves;

   a. Verified open at least once per month.

   b. Test as specified in 3.13. These valves may be closed intermittently for testing under administrative control.

2. During each refueling interval verify the scram discharge volume drain and vent valves;

   a) Close within 30 seconds after receipt of a reactor scram signal and

   b) Open when the scram is reset.

BASES:

3/4.3 REACTIVITY CONTROL

A. Reactivity Limitation

1. The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.25% Δk at the time of the test, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in % Δk is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., the initial loading. R must be a positive quantity or zero. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter.

   The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all inverted poison tubes present in the core. A new value of R must be determined for each full cycle.

   The 0.25% Δk in the expression R + 0.25% Δk is provided as a finite, demonstrable, subcriticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.25% Δk in reactivity, or by an insequence, xenon-free cold critical measurement to demonstrate at least R + 0.25% Δk in reactivity with the most reactive control rod fully withdrawn. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but at least an R + 0.25% Δk margin beyond this.

BASES:

3/4.3 REACTIVITY CONTROL (Cont)

A. Reactivity Limitation (Cont)

2. Reactivity margin - Inoperable control rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and then disarmed electrically*, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shutdown at all times with the remaining control rods assuming the strongest operable control rod does not insert. An allowable pattern for control rods valved out of service, which shall meet this Specification, will be determined and made available to the operator. The number of rods permitted to be inoperable could be many more than the eight allowed by the Specification, particularly late in the operation cycle; however, the occurrence of more than eight could be indicative of a generic control rod drive problem and the reactor will be shut down. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propogated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition.

* To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out of the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication.

B. Control Rod Withdrawal (Cont)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given is subsection 3.5.2 of the FSAR, and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. In the course of performing normal startup and shutdown procedures, the reactor operator follows a pre-specified sequence for the withdrawal or insertion of control rods. The specified sequences are characterized by homogeneous, scattered patterns of control rods selected for withdrawal or insertion. The maximum control rod worths encountered in these patterns for the initial core load are presented in FSAR Figure R.3-1. These sequences are developed prior to initial operation of the unit to limit the reactivity worths of individual control rods in the core. These control rod sequences, together with the integral rod velocity limiters which will limit the velocity during free fall to less than five feet per second, limit the potential reactivity insertion such that the consequences of a control rod drop accident will not exceed a peak calculated enthalpy of 280 calories/gram generated in the fuel. The design limit of 280 calories/gram is selected for limiting peak enthalpies in UO_2 and is assumed to be the lower threshold at which rapid fuel dispersal and damaging pressure pulses to the primary system might occur.

   As discussed in FSAR Section 14.5.1.3, the calculated radiological consequences of a control rod drop accident are well within the guideline values of 10 CFR Part 100.

   When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

B. Control Rod Withdrawal (Cont)

We are therefore requiring as a limiting condition of operation (LCO) that the Rod Worth Minimizer (RWM) be operable when the reactor is critical and below 20% of design power in accordance with Specification 3.3.B.3a so that the maximum in-sequence control rod worth will be limited to 0.010 delta k as given in Specification 3.3.B.3b(1) even assuming a single failure of the RWM or an operator error. The RWM assists and supplements the operator with an effective backup control rod monitoring routine that enforces adherence to pre-established startup, shutdown, and low power level control rod procedures. The RWM computer prevents the operator from establishing control rod patterns that are not consistent with prestored RWM sequences by initiating appropriate rod select block, rod withdrawal block, and rod insert block - interlock signals to the reactor manual control systems rod block circuitry. Reference: FSAR Section 7.16.4.3. The RWM sequences stored in the computer memory are based on control rod withdrawal procedures designed to limit the individual control rod worths to levels given in Specification 3.3.B.3.b.

Two exceptions to the requirement for RWM operability are permitted. Control rods may be moved to shutdown the reactor, and up to two control rods can be moved provided all other rods, except those which cannot be moved with control rod drive pressure, are inserted. The first exception permits the operator to shutdown the reactor in the event the RWM should become inoperable while the reactor is critical. In this case, the operator is moving the rods to reduce the reactivity in the core. Outward movement of any control rod is limited to a short adjustment and the general sequence of control rod movement is always toward a safer pattern during shutdown operations. The second exception permits the control rod drives to be moved when the RWM is inoperative provided that all but two rods are fully inserted except for those control rods which cannot be moved with control rod drive pressure.

Above 10% of design power assuming a single operator error, it will not be possible for the maximum rod worth to exceed 0.020 delta K in accordance with Specification 3.3.B.3.b(2).

Specification 4.3.B.3 requires a sequence of checks and tests on the RWM to verify its operability before startup and before reducing power to less than 10% of design power. These checks and tests assure that the actions of the control operator are always monitored and blocked when in error should they lead to a condition which might cause fuel damage during the control rod drop accident.

B. Control Rod Withdrawal (Cont)

Under these specification limits, the maximum energy deposition in the fuel and the number of fuel rods damaged resulting from a control rod drop accident, assuming Technical Specification limits on scram times (Specification 3.3.C) and rod drop velocity (5 feet/second), is established to be below the consequences calculated by the licensee for the hot standby critical case. Reference: FSAR Section 14.5.1. Therefore, the assumptions used by the licensee and the NRC in estimating the number of failed fuel rods and fuel damage resulting from the excursion energy generated by the rod drop accident appear conservative within the LCO.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to critical using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal. Two channels are provided, of which one may be bypassed for not more than 24 hours without significantly increasing the risk from a rod withdrawal error. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences.

   During reactor operation with a limiting control rod pattern, the reactor in run and reactor power greater than 29% of rated power, RBM operability is required to prevent the withdrawal of a single control rod which could result in MCPR being decreased below the MCPR safety limit. A limiting control rod pattern exists when MCPR is below a threshold MCPR which does not provide adequate margin to the safety limit MCPR in the event a rod is fully withdrawn. When RBM operability is required and one RBM channel is inoperable, an instrument functional test of the operable RBM channel prior to control rod withdrawal will provide adequate assurance that improper withdrawal does not occur.

BASES:

3/4.3 REACTIVITY CONTROL (Cont)

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit MCPR. Analysis of the limiting power transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than the Safety Limit MCPR.

The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be scram tested during each cycle as a periodic check against deterioration of the control rod performance.

In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.3.C.

D. Control Rod Accumulators

Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a k_{eff} <1.0 - other repeating rod sequences with more rods withdrawn resulted in k_{eff} >1.0. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% ΔK. Deviations in core reactivity greater than 1% ΔK are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System.

Objective:

To assure the availability of a system with the capability to shut down the reactor and maintain the shut down condition without the use of control rods.

Specification:

A. Normal System Availability

1. During periods when fuel is in the reactor and prior to startup from a Cold Condition, the Standby Liquid Control System shall be operable, except as specified in 3.4.B below. This system need not be operable when the reactor is in the Cold Shutdown Condition, all operable control rods are fully inserted and Specification 3.3.A is met.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. When tested as specified in 3.13 verify that each pump delivers at least 39 GPM against a system head of 1275 psig.

2. As required below:

   a. Once every refueling interval while testing as specified in 3.13 verify the system relief valve set point of 1425 psig ± 43 psig.

3.4 STANDBY LIQUID CONTROL SYSTEM (Cont)

B. Operation with Inoperable Components:

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

4.4 STANDBY LIQUID CONTROL SYSTEM (Cont)

A. Normal System Availability (Cont)

b. At least once during each refueling interval, while testing as specified in 3.13, manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.

This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump operability. The replacement charges to be installed will be selected from the same manufactured batch as the tested charge.

c. When testing to satisfy requirement 4.4.A.2.b, both systems, including both explosive valves, shall be tested in the course of two refueling intervals.

B. Surveillance with Inoperable Components:

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM (Cont)

C. Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:

1. The net volume - concentration of the Liquid Control Solution in the liquid control tank shall be maintained as required in Figure 3.4-1.

2. The temperature of the liquid control solution shall be maintained above 48°F. If the solution temperature falls to or below 48°F, the system will be flow tested to verify a flow path.

3. The enrichment of the liquid control solution shall be maintained at a B^{10} isotope enrichment exceeding 54.5 atom percent.

D. There are two operational considerations associated with the Standby Liquid Control sodium pentaborate solution requirements. The first consideration involves sodium pentaborate concentration/ volume requirements. The second consideration involves B^{10} isotopic enrichment. The related Limiting Conditions for operation are delineated below:

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM (Cont)

C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Temperature: Check at least once per day.

3. Concentration: Check at least once per month. Also check concentration anytime water or boron is added to the solution, or the solution is at or below 48°F.

4. Enrichment: Check B^{10} enrichment level by test anytime boron is added to the solution and during each refueling outage. Enrichment analyses shall be received within 30 days of test performance. If not received within 30 days, see Table 6.9.1 for reporting requirements.

3.4 STANDBY LIQUID CONTROL SYSTEM (Cont)

D. (Cont)

1. If specification 3.4.A, B, or C.1 or C.2 cannot be met, the reactor shall be placed in a Cold Shutdown Condition with all operable control rods fully inserted within 24 hours.

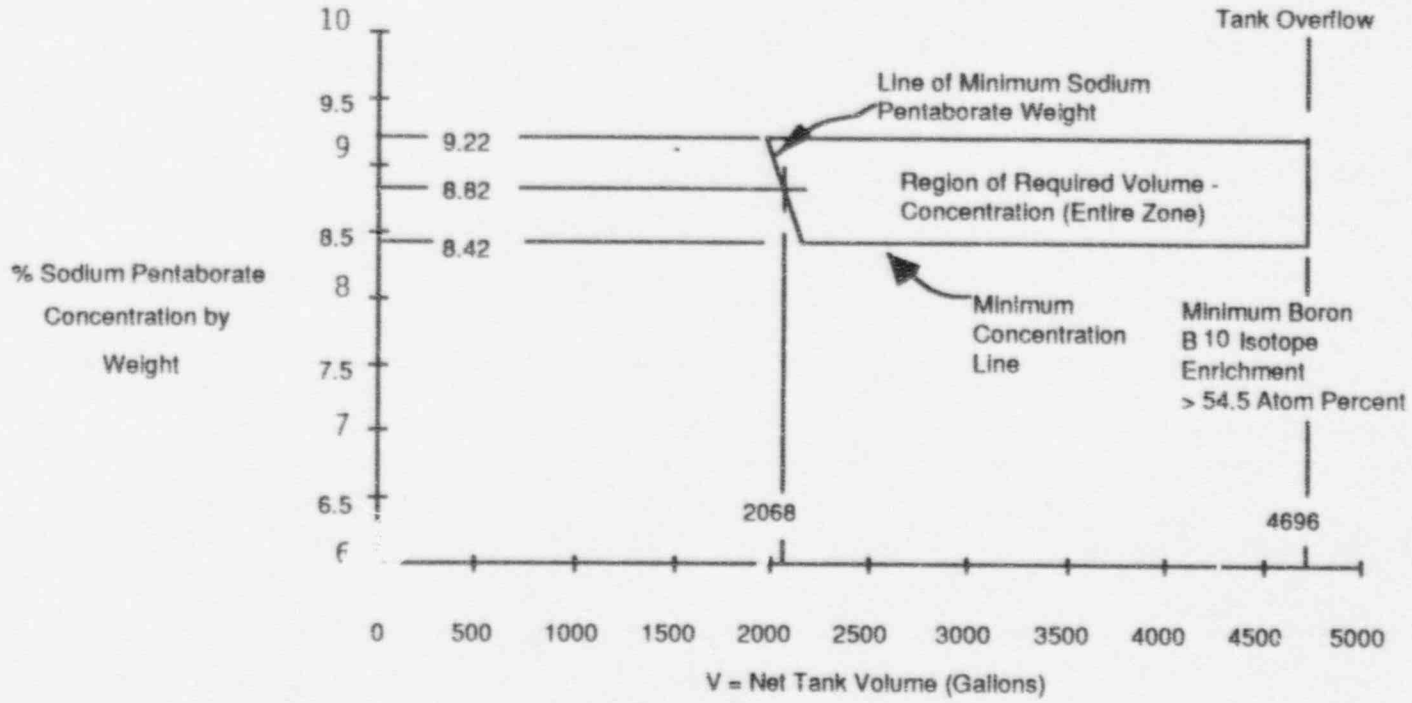
2. If the enrichment requirements of specification 3.4.C.3 are not met, a check shall be made to ensure that sodium pentaborate solution meets the original design criteria by comparing the enrichment, concentration and volume to established criteria. If the sodium pentaborate solution does not meet the original design criteria, the reactor shall be placed in a Cold shutdown Condition with all operable control rods fully inserted within 24 hours.

3. If the Sodium pentaborate solution meets the original design criteria, but the enrichment requirements of specification 3.4.C.3 are not met, bring the B^{10} isotopic enrichment to greater than 54.5 atom percent within seven days from the time of receipt of the enrichment report. If after this time period the enrichment requirements of specification 3.4.C.3 are still not met, submit a report to the NRC within seven days and advise them of plans to bring the solution up to a demonstratable 54.5 atom percent B^{10} isotopic enrichment.

4.4 STANDBY LIQUID CONTROL SYSTEM (Cont)

PNPS
Figure 3.4 - 1

Sodium Pentaborate Solution
Volume and Concentration Requirements



Revision 177
Amendment ~~102~~

BASES:

3/4.4 STANDBY LIQUID CONTROL SYSTEM

A. The requirements for SLC capability to shutdown the reactor are identified via the station Nuclear Safety Operational Analysis (Appendix G to the FSAR, Special Event 45). If no more than one operable control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the Standby Liquid Control system is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the standby liquid control system is required. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the Standby Liquid Control system is designed to inject a quantity of boron that produces a minimum concentration equivalent to 675 ppm of natural boron in the reactor core. The 675 ppm equivalent concentration in the reactor core is required to bring the reactor from full power to at least a three percent Δk subcritical condition, considering the hot to cold reactivity difference, xenon poisoning etc. The system will inject this boron solution in less than 125 minutes. The maximum time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

The Standby Liquid Control system is also required to meet 10CFR50.62 (Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants). The Standby Liquid Control system must have the equivalent control capacity (injection rate) of 86 gpm at 13 percent by wt. natural sodium pentaborate for a 251" diameter reactor pressure vessel in order to satisfy 10CFR50.62 requirements. This equivalency requirement is fulfilled by a combination of concentration, B^{10} enrichment and flow rate of sodium pentaborate solution. A minimum 8.42% concentration and 54.5% enrichment of B^{10} isotope at a 39 GPM pump flow rate satisfies the ATWS Rule (10CFR50.62) equivalency requirement.

Because the concentration/volume curve has been revised to reflect the increased B^{10} isotopic enrichment, an additional requirement has been added to evaluate the solution's capability to meet the original design shutdown criteria whenever the B^{10} enrichment requirement is not met.

Testing the pumps and valves in accordance with ASME B&PV Code Section XI (Articles IWP and IWV, except where specific relief is granted) adequately assesses component operational readiness. The only practical time to fully test the liquid control system is during a refueling outage. Various components of the system are individually tested periodically, thus making more frequent testing of the entire system unnecessary.

BASES:

3/4.4 STANDBY LIQUID CONTROL SYSTEM (Cont)

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten the shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the remaining system will perform its intended function and that the long term average availability of the system is not reduced is obtained for a one out of two system by an allowable equipment out of service time of one third of the normal surveillance frequency. This method determines an equipment out of service time of ten days. Additional conservatism is introduced by reducing the allowable out of service time to seven days, and by increased testing of the operable redundant component.

C. The quantity of B^{10} stored in the Standby Liquid Control System Storage Tank is sufficient to bring the concentration of B^{10} in the reactor to the point where the reactor will be shutdown and to provide a minimum 25 percent margin beyond the amount needed to shutdown the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. Test intervals for level monitoring have been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution shall be kept at least 10°F above the maximum saturation temperature to guard against boron precipitation. Minimum solution temperature is 48°F. This is 10°F above the saturation temperature for the maximum allowed sodium pentaborate concentration of 9.22 Wt. Percent.

Each parameter (concentration, pump flow rate, and enrichment) is tested at an interval consistent with the potential for that parameter to vary and also to assure proper equipment performance. Enrichment testing is required when material is received and when chemical addition occurs since change cannot occur by any process other than the addition of new chemicals to the Standby Liquid Control solution tank.

LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and suppression pool cooling systems.

Objective

To assure the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Systems

1. Both core spray systems shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.2 below.

SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the Surveillance Requirements of the core and suppression pool cooling systems which are required when the corresponding Limiting Condition for operation is in effect.

Objective

To verify the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Systems

1. Core Spray System Testing.

Item	Frequency
a. Simulated Automatic Actuation test.	Once/ Operating Cycle
b. Pump Operability	When tested as specified in 3.13 verify that each core spray pump delivers at least 3300 GPM against a system head corresponding to a reactor vessel pressure of 104 psig

LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

A. Core Spray and LPCI Systems (Cont)

2. From and after the date that one of the core spray systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days, provided that during such seven days all active components of the other core spray system and active components of the LPCI system and the diesel generators are operable.

3. The LPCI system shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.4 and 3.5.F.5.

4. From and after the date that the LPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during such seven days the containment cooling system (including 2 LPCI pumps) and active components of both core spray systems, and the diesel generators required for operation of such components if no external source of power were available shall be operable.

5. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

A. Core Spray and LPCI Systems (Cont)

c. Motor Operated Valve Operability	As specified in 3.13
d. Core Spray Header Δp Instrumentation	
Check	Once/day
Calibrate	Once/3 months
Test Step	Once/3 months

2. This section intentionally left blank

3. LPCI system Testing shall be as follows:

a. Simulated Automatic Actuation Test	Once/ Operating Cycle
b. Pump Operability	When tested as specified in 3.13 verify that each LPCI pump delivers 4800 GPM at a head across the pump of at least 380 ft
c. Motor Operated valve operability	As specified in 3.13

LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

B. Containment Cooling System

1. Except as specified in 3.5.B.2 and 3.5.F.3 below, both containment cooling system loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, and prior to reactor startup from a Cold Condition.

2. From and after the date that one containment cooling system loop is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 72 hours unless such system loop is sooner made operable, provided that the other containment cooling system loop, including its associated diesel generator, is operable.

3. If the requirements of 3.5.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

B. Containment Cooling System

1. Containment Cooling system Testing shall be as follows.

Item	Frequency
a. Pump Operability	When tested as specified in 3.13 verify that each RBCCW pump delivers 1700 GPM at 70 ft TDH and each SSW pump delivers 2700 GPM at 55 ft TDH
b. Valve Operability	As specified in 3.13
c. Air test on drywell and torus headers and nozzles	Once/5 years

LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

C. HPCI System

1. The HPCI system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and reactor coolant temperature is greater than 365°F; except as specified in 3.5.C.2 below.

2. From and after the date that the HPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days all active components of the ADS system, the RCIC system, the LPCI system and both core spray systems are operable.

3. If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to or below 150 psig within 24 hours.

SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

C. HPCI System

1. HPCI system testing shall be performed as follows:

a. Simulated Automatic Actuation Test	Once/ operating cycle
b. Pump Operability	When tested as specified in 3.13 verify that the HPCI pump delivers at least 4250 GPM for a system head corresponding to a reactor pressure of 1000 psig
c. Motor Operated Valve Operability	As specified in 3.13
d. Flow Rate at 150 psig	Once/ operating cycle verify that the HPCI pump delivers at least 4250 GPM for a system head corresponding to a reactor pressure of 150 psig

The HPCI pump shall deliver at least 4250 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

D. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC system shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and reactor coolant temperature is greater than 365°F; except as specified in 3.5.D.2 below.

2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding 14 days provided that during such 14 days the HPCIS is operable.

3. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to or below 150 psig within 24 hours.

SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

D. Reactor Core Isolation Cooling (RCIC) System

1. RCIC system testing shall be performed as follows:

a. Simulated Automatic Actuation Test	Once/operating cycle
b. Pump Operability	When tested as specified in 3.13 verify that the RCIC pump delivers at least 400 GPM at a system head corresponding to a reactor pressure of 1000 psig
c. Motor Operated Valve Operability	As specified in 3.13
d. Flow Rate at 150 psig	Once/operating cycle verify that the RCIC pump delivers at least 400 GPM at a system head corresponding to a reactor pressure of 150 psig

The RCIC pump shall deliver at least 400 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

E. Automatic Depressurization System (ADS)

1. The Automatic Depressurization System shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 104 psig and prior to a startup from a Cold Condition, except as specified in 3.5.E.2 below.

2. From and after the date that one valve in the Automatic Depressurization System is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such valve is sooner made operable, provided that during such 14 days the HPCI system is operable.

3. If the requirements of 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to at least 104 psig within 24 hours.

SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

E. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

   a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. The ADS manual inhibit switch will be included in this test.

   b. With the reactor at pressure, each relief valve shall be manually opened until a corresponding change in reactor pressure or main turbine bypass valve positions indicate that steam is flowing from the valve.

LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

F. Minimum Low Pressure Cooling and Diesel Generator Availability

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling systems and the remaining diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown condition, both core spray systems, the LPCI and containment cooling systems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.

4. During a refueling outage, for a period of 30 days, refueling operation may continue provided that one core spray system or the LPCI system is operable or Specification 3.5.F.5 is met.

SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

F. Minimum Low Pressure Cooling and Diesel Generator Availability

1. When it is determined that one diesel generator is inoperable, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter until the inoperable diesel is repaired.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

F. Minimum Low Pressure Cooling and Diesel Generator Availability (Cont)

5. When irradiated fuel is in the reactor vessel and the reactor is in the Refueling Condition with the torus drained, a single control rod drive mechanism may be removed, if both of the following conditions are satisfied:

a) No work on the reactor vessel, in addition to CRD removal, will be performed which has the potential for exceeding the maximum leak rate from a single control blade seal if it became unseated.

b) i) the core spray systems are operable and aligned with a suction path from the condensate storage tanks.
ii) the condensate storage tanks shall contain at least 200,000 gallons of usable water and the refueling cavity and dryer/separator pool shall be flooded to a least elevation 114'-0"

G. Deleted

H. Maintenance of Filled Discharge Pipe

Whenever core spray systems, LPCI system, HPCI or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to, to assure that the discharge piping of the core spray systems, LPCI system, HPCI and RCIC are filled:

1. Every month the LPCI system and core spray system discharge piping shall be vented from the high point and water flow observed.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

4.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

H. Maintenance of Filled Discharge Pipe (Cont)

2. Following any period where the LPCI system or core spray systems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

4. The pressure switches which monitor the discharge lines to ensure that they are full shall be functionally tested every month and calibrated every three months.

BASES:

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray and LPCI System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss of coolant analysis performed by General Electric in accordance with Section 50.46 and Appendix K of 10CFR50, the Pilgrim I Emergency Core Cooling Systems are adequate to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit calculated fuel clad temperature to less than 2200°F, to limit calculated local metal water reaction to less than or equal to 17%, and to limit calculated core wide metal water reaction to less than or equal to 1%. The detailed bases is described in NEDC-31852P and summarized in Section 6.5 of the PNPS FSAR.

The analyses discussed in NEDC-31852P calculated a peak clad fuel temperature of less than 2200°F with a Core Spray pump flow of 3200 gallons per minute (gpm). A flow rate of 3300 gpm ensures adequate flow for events involving degraded voltage.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Pilgrim, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis takes credit for core spray flow into the core at vessel pressure below 205 psig. However, the analysis is conservative in that no credit is taken for spray cooling heat transfer in the hottest fuel bundle until the pressure at rated flow for the core spray (104 psig vessel pressure) is reached.

The LPCI system is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI system and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high pressure emergency core cooling systems. The analyses in NEDC-31852P calculated a peak clad fuel temperature of less than 2200°F with LPCI pump flows of 4550 gpm, 4033 gpm, and 3450 gpm for two, three, and four pump combinations feeding into a single loop. A single pump flow rate at 4800 gpm ensures sufficient flow to meet or exceed the analyses' assumptions.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

The analyses of LOCA for PNPS demonstrated the combination of LPCS/LPCI systems are sufficient to provide core cooling even with a single failure of either an active or passive safety-related component. The analyses determined there were four significant single failures that challenge the Emergency Core Coolant Systems' capability to prevent fuel damage during the postulated LOCA. They are:

1) Battery Failure - Loss of a single battery train could leave only one LPCS pump, two LPCI pumps, and ADS to mitigate the LOCA. This is the most limiting single failure for all but the largest postulated recirculation line breaks and for all postulated non-recirculation line breaks.

2) LPCI Injection Valve Failure - Loss of the injection valve selected by LPCI Loop Selection Logic for the pathway for all LPCI pumps' flow leaves two core spray pumps, HPCI, and ADS for LOCA mitigation. This becomes the limiting single failure for the largest postulated recirculation line breaks.

3) Loss of one emergency diesel generator - This leaves one LPCS pump, two LPCI pumps, and ADS for LOCA mitigation.

4) HPCI Failure - This leaves all other ECCS resources available. It is a significant failure primarily for small line breaks.

In all cases above, the remaining ECCS resources are sufficient to prevent PCT from exceeding 2200°F and other criteria provided in Section 50.46 and Appendix K of 10CFR50.

Each Core Spray system consists of one pump and associated piping and valves with all active components required to be operable. The LPCI system consists of four LPCI pumps and associated piping and valves with all active components required to be operable.

Should one Core Spray system become inoperable, the remaining Core Spray and the LPCI system are available should the need for core cooling arise. Based on judgments of the reliability of the remaining systems (i.e., the Core Spray and LPCI) a seven-day repair period was obtained.

If the LPCI system is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. Based on judgments of the reliability of the remaining Core Spray systems, a 7-day repair period was set.

The LPCI system is not considered inoperable when the RHR System is operating in the shutdown cooling mode.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

B. Containment Cooling System

The containment cooling system for Pilgrim I consists of two independent loops, each of which to be an operable loop requires one LPCI pump, two RBCCW pumps, and two SSW pumps be operable. There are installed spares for margin above the design conditions. Each system has the capability to perform its function; i.e., removing 64 x 10^6 Btu/hr (Ref. Amendment 18), even with some system degradation. If one loop is out-of-service, reactor operation is permitted for 72 hours.

With components or systems out-of-service, overall core and containment cooling reliability is maintained by the operability of the remaining cooling equipment.

Since some of the SSW and RBCCW pumps are required for normal operation, capacity testing of individual pumps by direct flow measurement is impractical. Pump operability will be demonstrated during normal system operation and/or when system conditions allow capacity and performance testing in accordance with 3.13.

C. HPCI

The limiting conditions for operating the HPCI System are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the HPCI System (FSAR Section 6).

HPCI is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss-of-coolant which does not result in rapid depressurization of the reactor vessel. HPCI permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. HPCI continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray System operation maintains core cooling.

The capacity of the system is selected to provide this required core cooling. The HPCI pump is designed to pump 4250 gpm at reactor pressures between 1100 and 150 psig. Two sources of water are available. Initially, demineralized water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI System. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reached equilibrium with the flow through the break. Continued depressurization causes the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

The analysis in FSAR Appendix G shows that the ADS provides a single failure proof path for depressurization for postulated transients and accidents. The RCIC is required as an alternate source of makeup to the HPCI only in the case of loss of all offsite A-C power. Considering the HPCI and the ADS plus RCIC as redundant paths, and considering judgments of the reliability of the ADS and RCIC systems, a 14-day allowable repair time is specified.

The requirement that HPCI be operable when reactor coolant temperature is greater that 365°F is included in Specification 3.5.C.1 to clarify that HPCI need not be operable during certain testing (e.g., reactor vessel hydro testing at high reactor pressure and low reactor coolant temperature). 365°F is approximately equal to the saturation steam temperature at 150 psig.

D. RCIC System

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the normal heat sink is unavailable. The Station Nuclear Safety Operational Analysis, FSAR Appendix G, shows that RCIC also serves as redundant makeup system on total loss of all offsite power in the event that HPCI is unavailable. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgments on the reliability of the HPCI system, an allowable repair time of 14 days is specified.

The requirement that RCIC be operable when reactor coolant temperature is greater than 365°F is included in Specification 3.5.D.1 to clarify that RCIC need not be operable during certain testing (e.g., reactor vessel hydro testing at high reactor pressure and low reactor coolant temperature). 365°F is approximately equal to the saturation steam temperature at 150 psig.

BASES:

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

E. Automatic Depressurization System (ADS)

The limiting conditions for operating the ADS are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the ADS (FSAR Section 6).

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray systems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI or Core Spray. There are four valves provided and each has a capacity of 800,000 lb/hr at a reactor pressure of 1125 psig.

The allowable out of service time for one ADS valve is determined as 14 days because of the redundancy and because of HPCI operability; therefore, redundant protection for the core with a small break in the nuclear system is still available.

The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

F. Minimum Low Pressure Cooling and Diesel Generator Availability

The purpose of Specification F is to assure that adequate core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only 2 LCPI pumps would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Specification F allows removal of one CRD mechanism while the torus is in a drained condition without compromising core cooling capability. The available core cooling capability for a potential draining of the reactor vessel while this work is performed is based on an estimated drain rate of 300 gpm if the control rod blade seal is unseated. Flooding the refuel cavity and dryer/separator pool to elevation 114'-0" corresponds to approximately 350,000 gallons of water and will provide core cooling capability in the event leakage from the control rod drive does

3.5 CORE AND CONTAINMENT COOLING SYSTEMS (Cont)

occur. A potential draining of the reactor vessel (via control rod blade leakage) would allow this water to enter into the torus and after approximately 140,000 gallons have accumulated (needed to meet minimum NPSH requirements for the LPCI and/or core spray pumps), the torus would be able to serve as a common suction header. This would allow a closed loop operation of the LPCI system and the core spray system (once re-aligned) to the torus. In addition, the other core spray system is lined up to the condensate storage tanks which can supplement the refuel cavity and dryer/separator pool water to provide core flooding, if required.

Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

G. Deleted

H. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI system, HPCI, and RCIC are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. An analysis has been done which shows that if a water hammer were to occur at the time at which the system were required, the system would still perform its design function. However, to minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS SURVEILLANCE FREQUENCIES

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated valves are tested in accordance with ASME B&PV Code, Section XI (IWP and IWV, except where specific relief is granted) to assure their operability. The frequency and methods of testing are described in the PNPS IST program. The PNPS IST Program is used to assess the operational readiness of pumps and valves that are safety-related or important to safety. When components are tested and found inoperable the impact on system operability is determined, and corrective action or Limiting Conditions of Operation are initiated. A simulated automatic actuation test once each cycle combined with code inservice testing of the pumps and valves is deemed to be adequate testing of these systems.

The surveillance requirements provide adequate assurance that the core and containment cooling systems will be operable when required.

LIMITING CONDITION FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period except when the vessel temperatures are above 450°F. The reactor vessel flange to adjacent reactor vessel shell temperature differential shall not exceed 145°F.

2. The reactor vessel shall not be pressurized for hydrostatic and/or leakage tests, and subcritical or critical core operation shall not be conducted unless the reactor vessel temperatures are above those defined by the appropriate curves on Figures 3.6-1, 3.6-2, and 3.6-3. (Linear interpolation between curves is permitted). At stated pressure, the reactor vessel bottom head may be maintained at temperatures below those temperatures corresponding to the adjacent reactor vessel shell as shown in Figures 3.6-1 and 3.6-2.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, with the reactor vessel temperature less than or equal to 450°F, the temperatures at the following locations shall be permanently logged at least every 15 minutes until the difference between any two readings at individual locations taken over a 45 minute period is less than 5°F:

   a. Reactor vessel shell adjacent to reactor vessel flange

   b. Reactor vessel shell flange

   c. Recirculation loops A and B

2. Reactor vessel shell temperatures, including reactor vessel bottom head, and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented.

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

A. Thermal and Pressurization Limitations (Cont)

In the event this requirement is not met, achieve stable reactor conditions with reactor vessel temperature above that defined by the appropriate curve and obtain an engineering evaluation to determine the appropriate course of action to take.

3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 55°F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

6. Thermal-Hydraulic Stability

Core thermal power shall not exceed 25% of rated thermal power without forced recirculation.

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

A. Thermal and Pressurization Limitations (Cont)

Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 Mev neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to the requirements of ASTM E 185-66. Selected neutron flux specimens shall be removed at the frequency required by Table 4.6-3 and tested to experimentally verify adjustments to Figures 3.6-1, 3.6-2, and 3.6-3 for predicted NDT temperature irradiation shifts.

3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

LIMITING CONDITION FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

B. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed 20 microcuries of total iodine per ml of water.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour, except as specified in 3.6.B.3:

   Conductivity.....2 μmho/cm
   Chloride ion.....0.1 ppm

3. For reactor startups and for the first 24 hours after placing the reactor in the power operating condition, the following limits shall not be exceeded.

   Conductivity.....10 μmho/cm
   Chloride ion.....0.1 ppm

4. Except as specified in 3.6.B.3 above, the reactor coolant water shall not exceed the following limits when operating with steaming rates greater than or equal to 100,000 pounds per hour.

   Conductivity.....10 μmho/cm
   Chloride ion.....1.0 ppm

5. If Specification 3.6.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in Hot Shutdown within 24 hrs. and Cold Shutdown within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

B. Coolant Chemistry

1. a. A reactor coolant sample shall be taken at least every 96 hours and analyzed for radioactivity content.

   b. Isotopic analysis of a reactor coolant sample shall be made at least once per month.

2. During startups and at steaming rates less than 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for chloride content.

3. a. With steaming rates of 100,000 pounds per hour or greater, a reactor coolant sample shall be taken at least every 96 hours and analyzed for chloride ion content.

   b. When all continuous conductivity monitors are inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

C. Coolant Leakage

Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following limits shall be observed:

1. Operational Leakage

a. Reactor coolant system leakage shall be limited to:

1. No Pressure Boundary Leakage

2. ≤5 gpm Unidentified Leakage

3. ≤25 gpm Total Leakage averaged over any 24 hour period.

4. ≤2 gpm increase in Unidentified Leakage within any 24 hour period when in RUN mode.

b. With any reactor coolant system leakage greater than the limits of 2. and/or 3., above, reduce the leakage to within acceptable limits within 4 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

c. With any reactor coolant system leakage greater than the limits of 4. above, identify the source of leakage within 4 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

C. Coolant Leakage

Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following surveillances shall be performed:

1. Operational Leakage

Demonstrate drywell leakage is within the limits specified in 3.6.C.1 by monitoring the coolant leakage detection systems required to be operable by 3.6.C.2 at least once every 8 hours.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

C. Coolant Leakage (Cont)

d. When any Pressure Boundary Leakage is detected be in at least Hot Shutdown within the next 12 hours and be in Cold Shutdown within the next 24 hours.

2. Leakage Detection Systems

a. The following reactor coolant system leakage detection systems shall be Operable:

1. One drywell sump monitoring system, and either

2. One channel of a drywell atmospheric particulate radioactivity monitoring system, or

3. One channel of a drywell atmospheric gaseous radioactivity monitoring system.

b. 1. At least one drywell sump monitoring system shall be Operable; otherwise, be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

2. At least one gaseous or particulate radioactivity monitoring channel must be Operable; otherwise, reactor operation may continue for up to 31 days provided grab samples are obtained and analyzed every 24 hours, or be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

C. Coolant Leakage (Cont)

2. Leakage Detection Systems

The following reactor coolant leakage detection systems shall be demonstrated Operable:

a. For each required drywell sump monitoring system perform:

1. An instrument functional test at least once per 31 days, and

2. An instrument channel calibration at least once per operating cycle.

b. For each required drywell atmospheric radioactivity monitoring system perform:

1. An instrument check at least once per day,

2. An instrument functional test at least once per 31 days, and

3. An instrument channel calibration at least once per operating cycle.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

c. With no required leakage detection systems Operable, be in Cold Shutdown within 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than 104 psig and temperature greater than 340°F, both safety valves and the safety modes of all relief valves shall be operable. The nominal setpoint for the relief/safety valves shall be selected between 1095 and 1115 psig. All relief/safety valves shall be set at this nominal setpoint ± 11 psi. The safety valves shall be set at 1240 psig ± 13 psi.

2. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be below 104 psig within 24 hours. Note: Technical Specifications 3.6.D.2 - 3.6.D.5 apply only when two Stage Target Rock SRVs are installed.

3. If the temperature of any safety relief discharge pipe exceeds 212°F during normal reactor power operation for a period of greater than 24 hours, an engineering evaluation shall be performed justifying continued operation for the corresponding temperature increases.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

D. Safety and Relief Valves

1. Testing of safety and relief/safety valves shall be in accordance with 3.13.

2. At least one of the relief/safety valves shall be disassembled and inspected each refueling outage.

3. Whenever the safety relief valves are required to be operable, the discharge pipe temperature of each safety relief valve shall be logged daily.

4. Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

D. Safety Relief Valves (Con't)

4. Any safety relief valve whose discharge pipe temperature exceeds 212°F for 24 hours or more shall be removed at the next cold shutdown of 72 hours or more, tested in the as-found condition, and recalibrated as necessary prior to reinstallation. Power operation shall not continue beyond 90 days from the initial discovery of discharge pipe temperatures in excess of 212°F for more than 24 hours without prior NRC approval of the engineering evaluation delineated in 3.6.D.3.

5. The limiting conditions of operation for the instrumentation that monitors tail pipe temperature are given in Table 3.2-F.

E. Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

E. Jet Pumps

Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously.

1. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

3. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from established jet pump delta P characteristics by more than 10%.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

F. Jet Pump Flow Mismatch

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than or equal to 80%.

2. If Specification 3.6.F.1 is exceeded immediate corrective action shall be taken. If recirculation pump speed mismatch is not corrected within 30 minutes, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours unless the recirculation pump speed mismatch is brought within limits sooner.

G. Structural Integrity

1. The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components," Articles IWA, IWB, IWC, IWD and IWF and mandatory appendices as required by 10CFR50.55a(g), except where specific relief has been granted by the NRC pursuant to 10CFR50.55a(g)(6)(i).

H. Deleted

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

F. Jet Pump Flow Mismatch

Recirculation pump speeds shall be checked and logged at least once per day.

G. Structural Integrity

Inservice inspection of components shall be performed in accordance with the PNPS Inservice Inspection Program. The results obtained from compliance with this program will be evaluated at the completion of each ten year interval. The conclusions of this evaluation will be reviewed with the NRC.

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

I. Shock Suppressors (Snubbers)

1. During all modes of operation except Cold Shutdown and Refuel, all safety-related snubbers listed in PNPS Procedures shall be operable except as noted in 3.6.I.2 through 3.6.I.3 below.

   An Inoperable Snubber is a properly fabricated, installed and sized snubber which cannot pass its functional test.

   Upon determination that a snubber is either improperly fabricated, installed or sized, the corrective action will be as specified for an inoperable snubber in Section 3.6.I.2.

2. From and after the time that a snubber is determined to be inoperable, replace or repair the snubber during the next 72 hours, and initiate an engineering evaluation to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubbers and to ensure that the supported component remains capable of meeting its intended function in the specific safety system involved.

   Further corrective action for this snubber, and all generically susceptible snubbers, shall be determined by an engineering evaluation.

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

I. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all safety related hydraulic and mechanical snubbers listed in PNPS Procedures.

The required visual inspection interval varies inversely with the observed cumulative number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original time interval has elapsed may not be used to lengthen the required interval.

Number of snubbers found inoperable during inspection or during inspection interval:

Inoperable Snubbers	Subsequent Visual Inspection Interval	
0	24 Months	± 25%
1	18 Months	± 25%
2	12 Months	± 25%
3,4	6 Months	± 25%
5,6,7	124 Days	± 25%
8,9	62 Days	± 25%
10 or more	31 Days	± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

I. Shock Suppressors (Snubbers) (Cont)

3. From and after the time a snubber is determined to be inoperable, improperly fabricated, improperly installed or improperly sized, if the requirements of Section(s) 3.6.I.1 and 3.6.I.2 cannot be met, then the affected safety system, or affected portions of that system, shall be declared inoperable, and the limiting condition for that system entered, as appropriate.

4. Snubbers may be added to, or removed from, per 10CFR50.59, safety related systems without prior NRC approval. The addition or deletion of snubbers shall be reported to the NRC in accordance with 10CFR50.59.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

I. Shock Suppressors (Snubbers) (Cont)

1. Visual Inspection Acceptance Criteria

A. Visual inspections shall verify:

1. That there are no visible indications of damage or impaired operability.

2. Attachments to the foundation or support structure are such that the functional capability of the snubber is not suspect.

B. Snubbers which appear INOPERABLE as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval provided that:

1. The cause of the rejection is clearly established and remedied for that particular snubber, and

2. The affected snubber is functionally tested, when necessary, in the as found condition and determined OPERABLE per specifications 4.6.I.2.B., 4.6.I.2.C., as applicable.

C. For any snubber determined inoperable per specification 4.6.I.2, clearly establish the cause of rejection and remedy the problem for that snubber, and any generically susceptible snubber.

2. Functional Tests (Hydraulic and Mechanical Snubbers)

A. Schedule

At least once per operating cycle, a representative sample (12.5% of the total of each type:

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

I. Shock Suppressors (Snubbers) (Cont)

hydraulic, mechanical) of snubbers in use in the plant shall be functionally tested, either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.6.I.2.B, or 4.6.I.2.C, as applicable, an additional 10% of that type of snubber shall be functionally tested.

B. General Snubber Functional Test Acceptance Criteria (Hydraulic and Mechanical)

The general snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber release, or bleedrate, as applicable, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

C. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

I. Shock Suppressors (Snubbers) (Cont)

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

3. Snubber Service Life Monitoring

A. A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained.

B. At least once per cycle, the installation and maintenance records for each safety related snubber listed in PNPS Procedures shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated, or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

C. This Snubber Service Life Monitoring Program shall become effective July 1, 1982.

Note: Tables 4.6-1 and 4.6-2 have been deleted.

PNPS
TABLE 4.6-3

REACTOR VESSEL MATERIAL
SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE

Capsule Number	Effective Full Power Years (EFPY)
1	4.17
2	15 (approx.)
3	32 (End of Life)

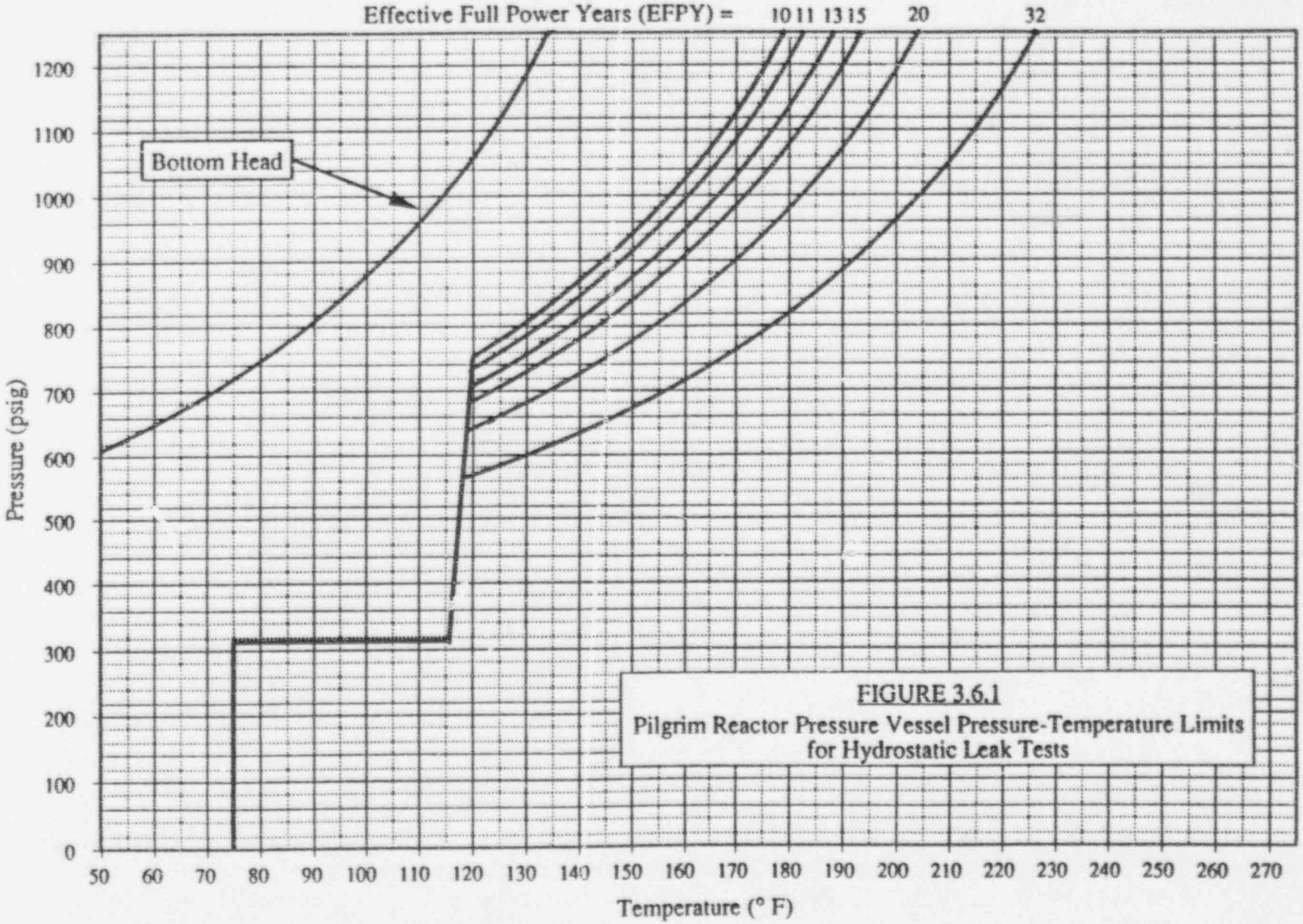


FIGURE 3.6.1
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Hydrostatic Leak Tests

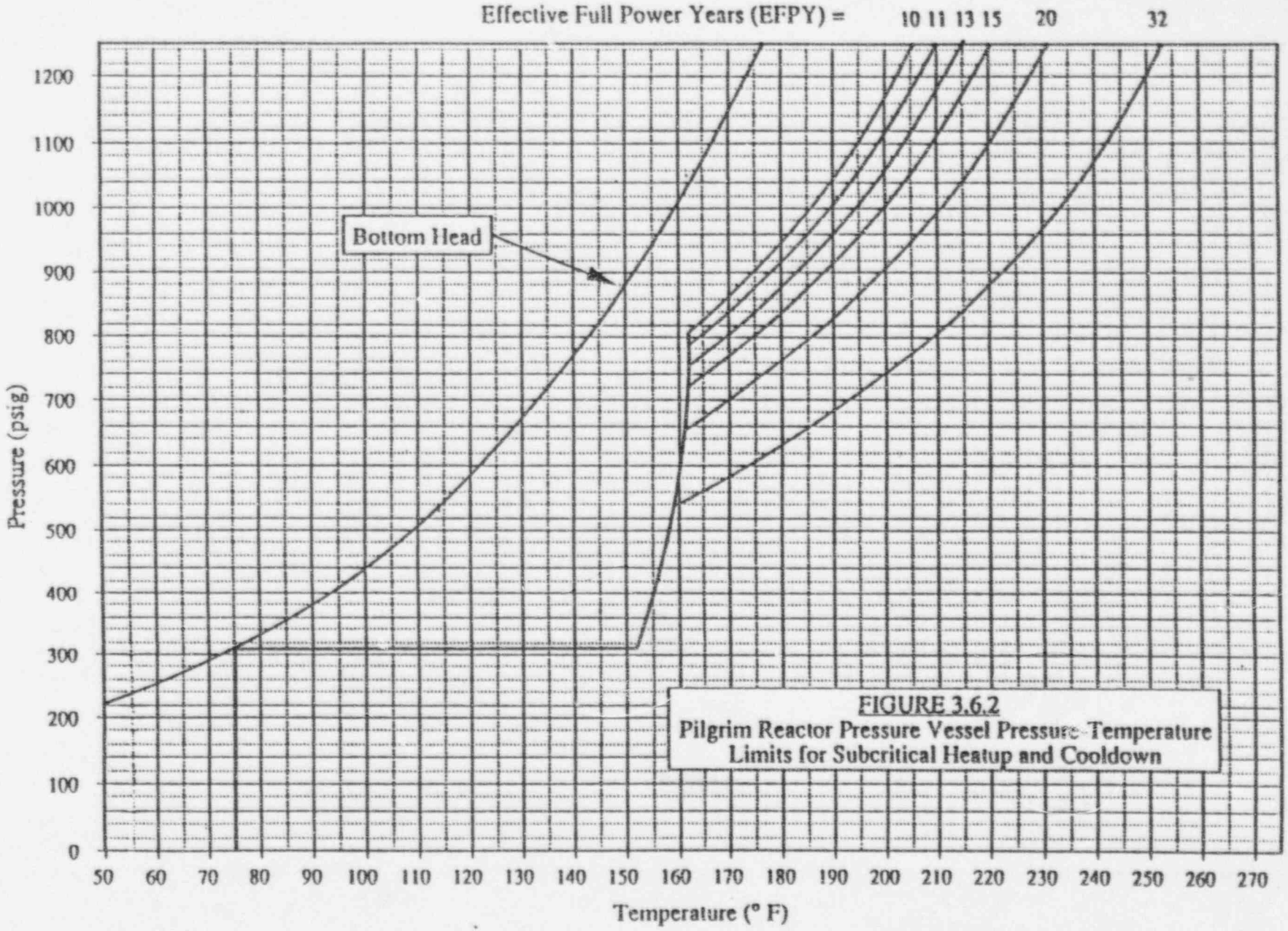


FIGURE 3.6.2
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Subcritical Heatup and Cooldown

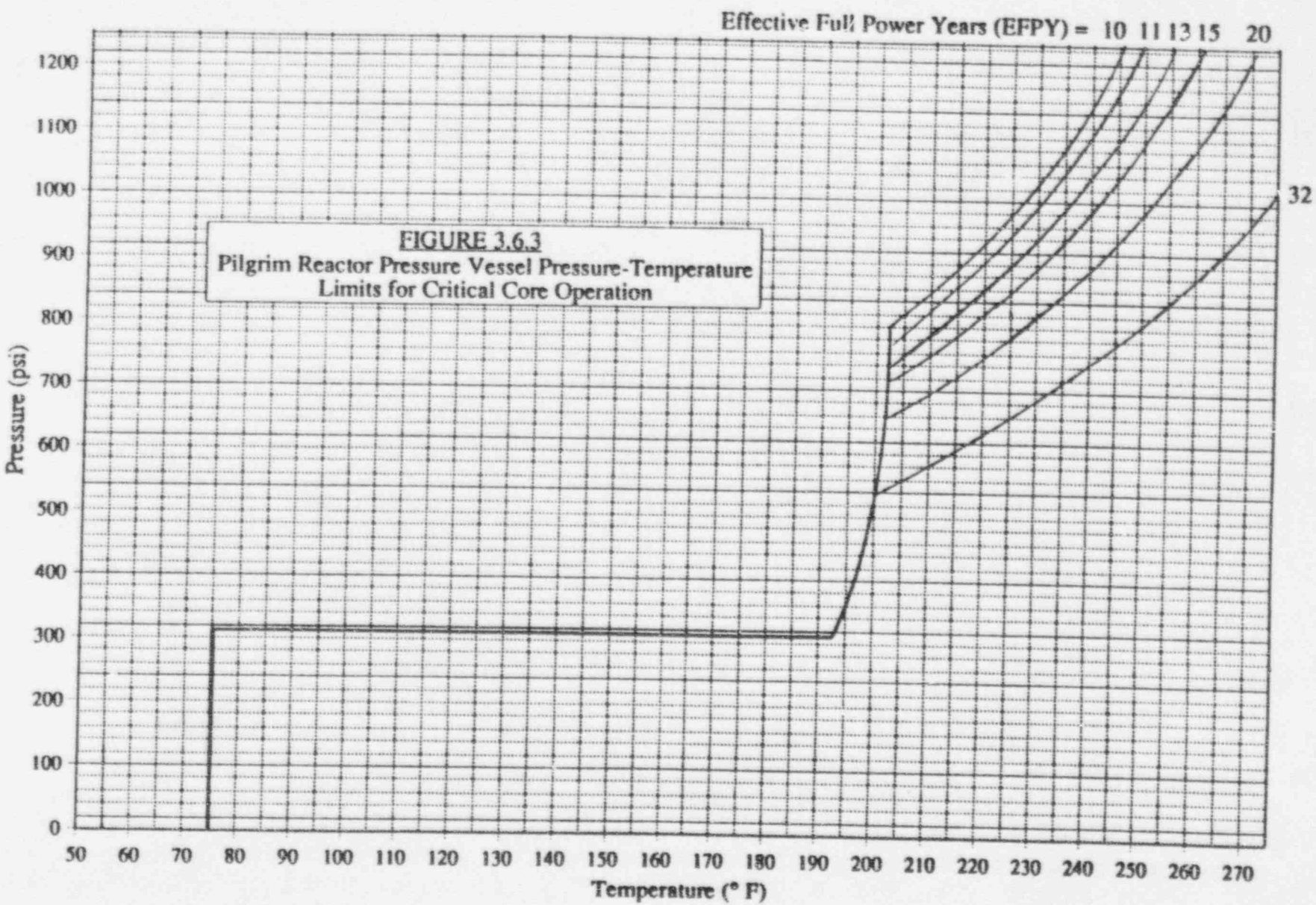


FIGURE 3.6.3
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Critical Core Operation

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY

A. Thermal and Pressurization Limitations

The allowable rate of heatup and cooldown for the reactor vessel contained fluid is 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 546°F. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

The manufacturer performed detailed stress analysis as shown in Amendment 17 of the FSAR. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

The permissible flange to adjacent shell temperature differential of 145°F is the maximum calculated for 100°F/hour heating and cooling rate applied continuously over a 100°F to 550°F range. The differential is due to the sluggish temperature response of the flange metal and its value decreases for any lower heating rate or the same rate applied over a narrower range.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result is stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

Appendix G to 10CFR50 defines the temperature-pressurization restrictions for hydrostatic and leak tests, pressurization, and critical operation. These limits have been calculated for Pilgrim and are contained in Figures 3.6-1, 3.6-2, and 3.6-3.

A. Thermal and Pressurization Limitations (Cont)

The bottom head, defined as the spherical portion of the reactor vessel located below the lower circumferential weld, was also evaluated. Reference transition temperatures (RT_{NDT}) were developed for the bottom head and the resulting pressure vs. temperature curves plotted on Figures 3.6-1 and 3.6-2. It has been determined that the bottom head temperatures are allowed to lag the vessel shell temperatures, Reference: Teledyne Engineering Services (TES) report TR-6051C-1, dated June 27, 1986. The referenced analysis utilizes the stress results established in the Combustion Engineering Inc., Pilgrim Reactor Vessel Design Report, No. CENC 1139, dated 1971, and combines the stress analysis results, specific to the bottom head, with the pressurization temperatures necessary to maintain fracture toughness requirements in accordance with the ASME Boiler and Pressure Vessel Code, Section III, the criteria of 10CFR50, Appendix G, and the supplementary guidelines of Reg. Guide 1.99, Rev. 2.

For Pilgrim pressure-temperature restrictions, two locations in the reactor vessel are limiting. The closure region controls at lower pressures and the beltline controls at higher pressures.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (>1 mev) above about 10^{17} nvt may shift the NDT temperature of the vessel metal above the initial value. Impact tests from the first material surveillance capsule removed at 4.17 EFPY indicated a maximum RT_{NDT} shift of 55°F for the weld specimens.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be periodically removed and tested to experimentally verify the values used for Figures 3.6-1, 3.6-2, and 3.6-3. The withdrawal schedule of Table 4.6-3 has been established as required by 10CFR50, Appendix H.

The RT_{NDT} of the closure region is - 5°F. The initial RT_{NDT} for the beltline weld and basemetal are -50°F and 0°F respectively. These RT_{NDT} temperatures are based upon unirradiated test data, adjusted for specimen orientation in accordance with USNRC Branch Technical Position MTEB 5-2.

The closure and bottom head regions are not exposed to neutron fluence (> 1 Mev) over the vessel life sufficient to cause a shift in RT_{NDT}. The pressure-temperature limitations (Figures 3.6-1, 3.6-2, and 3.6-3) of the closure and bottom head regions will therefore remain constant throughout vessel life. Only the beltline region of the reactor vessel will experience a shift in RT_{NDT} with a resultant increase in Pressure-Temperature limits.

The curves apply to 100% bolt preload condition, but are conservative for lesser bolt preload conditions.

A. Thermal and Pressurization Limitations (Cont)

For critical core operation when the water level is within the normal range for power operation and the pressure is less than 20% of the preservice system hydrostatic test pressure (313 psi), the minimum permissible temperature of the highly stressed regions of the closure flange is RT_{NDT} + 60 = 55°F. A conservative cutoff limit of 75°F was chosen as shown on Figure 3.6-3 and as permitted by 10CFR50 Appendix G, paragraph IV. A.3. This same cutoff is included in the limits for hydrostatic and leak tests and for non-critical operation, as shown on Figures 3.6-1 and 3.6-2 respectively, in order to be consistant with the limits for critical operation.

The closure region is more limiting than the feedwater nozzle with regards to both stress intensity and RT_{NDT}. Therefore the pressure-temperature limits of the closure are controlling.

The adjusted reference temperature shift is based on Regulatory Guide 1.99, Revision 2, dated May 1988; the analytical results of General Electric Report MDE 277-1285, Revision 1, dated January 21, 1985, regarding projected neutron fluence; and Teledyne Engineering Services Reports, TR-6052B-1, Revision 1, dated June 26, 1986, as supplemented by TR-7487, dated April 16, 1991, for RT_{NDT} vs. fluence as a function of temperature and pressure, and TR-6052C-1, dated June 27, 1986, for the RPV bottom head pressurization temperatures.

B. Coolant Chemistry

The reactor vessel coolant chemistry requirements are discussed in Subsection 4.2 of the FSAR.

A radioactivity concentration of 20 μ Ci/ml total iodine can be reached if there is significant fuel failure or if there is a failure or a prolonged shutdown of the cleanup demineralizer. Calculations performed by the AEC staff for this activity level results in a radiological dose at the site boundary of 8 rem to the thyroid from a postulated rupture of a main steam line assuming a 5 second valve closing time and a coolant inventory release of 3×10^4 lbs.

A reactor sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded.

B. Coolant Chemistry (Cont)

Materials in the primary system are primarily stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and given an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel. According to test data, allowable chloride concentrations could be set several orders of magnitude above the established limit at the oxygen concentration (.2-.3 ppm) experienced during power operation without causing significant failures. Zircaloy does not exhibit similar stress corrosion failures. However, there are some conditions under which the dissolved oxygen content of the reactor coolant water could be higher than .2-.3 ppm, such as refueling, reactor startup and hot standby. During these periods, a more restrictive limit of 0.1 ppm has been established to assure that permissible chloride-oxygen combinations are not exceeded. Boiling occurs at higher steaming rates causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels and assuring that the chloride-oxygen content is not such as would tend to induce stress corrosion cracking.

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the higher limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so high concentrations of chlorides are not considered harmful during these periods.

In the case of BWR's where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor clean-up system, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the clean-up system to re-establish the purity of the reactor coolant. During start-up periods, which are in the category of less than 1% reactor power, conductivity may exceed 2 μ mho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 μ mho/cm (other than short term spikes), samples will be taken to assure that the chloride concentration is less than 0.1 ppm.

B. Coolant Chemistry (Cont)

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meters inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The iodine radioactivity will be monitored by reactor water sample analysis. The total iodine activity would not be expected to change over a period of 96 hours. In addition, the trend of the stack off-gas release rate, which is continuously monitored, is an indication of the trend of the iodine activity in the reactor coolant. Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam lines.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses to determine major contributors to activity can be performed by a gamma scan.

C. Coolant Leakage

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

Verification of the integrity of the reactor coolant system (3.6.C.1.a.1: No Pressure Boundary Leakage) is provided during RPV Class I system hydrostatic and leak tests conducted to meet section 3/4.6.G: Structural Integrity (ASME Code, Section XI, IWA 5000, and IWB 5000.)

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from pump seal leakoffs, reactor vessel head flange seal leakoff, selected valve stem leakoff including recirculation loop and main steam isolation valves, and other equipment drains to the drywell equipment drain sump. The second sump, the drywell floor drain collection sump receives leakage from the drywell coolers, control rod drives, other valve stems and flanges, floor drains, and closed cooling water system drains. Drainage into the drywell floor drain sump is generally considered Unidentified Leakage. Both sumps are equipped with level and flow monitoring equipment to alert operators if allowable leak rates are approached.

A drywell sump monitoring system, as required in 3.6.C.2, consists of one equipment sump pump and one floor drain sump pump, plus associated instrumentation. Flow integrators, one for the equipment drain sump and

C. Coolant Leakage (Cont)

another for the floor sump, comprise the basic instrument system, and are used to record the flow of liquid from the drywell sumps. A manual system whereby the time interval between sump pump starts is utilized to provide a back-up to the flow integrators if the instrumentation is found to be inoperable. This time interval determines the leakage flow because the capacity of the pump is known.

The capacity of each of the two drywell floor sump pumps is 50 gpm and the capacity of each of the two drywell equipment sump pumps is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

In addition to the sump monitoring of coolant leakage, airborne radioactivity levels of the drywell atmosphere is monitored by the Reactor Pressure Boundary Leak Detection System. This system consists of two panels capable of monitoring the primary containment atmosphere for particulate and gaseous radioactivity as a result of coolant leaks.

The 2 gpm limit for coolant leakage rate increase within any 24 hour period is a limit specified by the NRC in Generic Letter 88-01: "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping". This limit applies only during the RUN mode to accommodate the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, which flows to the drywell equipment drain sump (Identified leakage) and floor drain sump (Unidentified leakage).

D. Safety and Relief Valves

The valve sizing analysis considered four, 10% capacity relief/safety valves and two 8% capacity safety valves. These sized and set pressures are established in accordance with the following three requirements of Section II of the ASME Code:

1. The lowest safety valve must be set to open at or below vessel design pressure and the highest safety valve be set at or below 105% of design pressure.

2. The valves must limit the reactor pressure to no more than 110% of design pressure.

3. Protection systems directly related to the valve sizing transient must not be credited with action (i.e., an indirect scram must be assumed).

D. Safety and Relief Valves (Cont)

A main steam line isolation with flux scram has been selected to be used as the safety valve sizing transient since this transient results in the highest peak vessel pressure of any transient when analyzed with an indirect scram. The original FSAR analysis concluded that the peak pressure transient with indirect scram would be caused by a loss of condenser vacuum (turbine trip with failure of the bypass valves to open). However, later observations have shown that the long lengths of steam lines to the turbine buffer the faster stop valve closure isolation and thereby reduce the peak pressure caused by this transient to a value below that produced by a main steam line isolation with flux scram.

Item 3 above indicates that no credit be taken for the primary scram signal generated by closure of the main steam isolation valves. Two other scram initiation signals would be generated, one due to high neutron flux and one due to high reactor pressure. Thus item 3 will be satisfied by assuming a scram due to high neutron flux.

Relieving capacity of 40% (4 relief/safety valves) results in a peak pressure during the transient conditions used in the safety valve sizing analysis which is well below the pressure safety limit.

The relief/safety valve settings satisfy the Code requirements that the lowest safety valve set point be at or below the vessel design pressure range to prevent unnecessary cycling caused by minor transients. The results of postulated transients where inherent relief/safety valve actuation is required are given in Appendices R and Q of the Final Safety Analysis Report.

Experience in safety valve operation shows that a testing of at least 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value of ± 1% is in accordance with Section III of the ASME Boiler and Pressure Vessel Code. An analysis has been performed which shows that with all safety valves set 1% higher, the reactor coolant pressure safety limit of 1375 psig is not exceeded.

The relief/safety valves have two functions; i.e., power relief or self-actuated by high pressure. Power relief is a solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate the valves to open. This function is discussed in Specification 3.5.D. In addition, the valves can be operated manually.

Pilgrim's experience with 2 stage safety/relief valves has demonstrated that minimum leakage exists when the tailpipe temperature is 215° Fahrenheit. Therefore, a reporting requirement triggered by a temperature of 212°F is conservative, and assures timely reporting before leakage reaches significant proportions.

E. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended recirculation line break. Therefore, if a failure occurred, repairs must be made.

A nozzle riser failure could cause the coincident failure of a jet pump body; however, because of the lack of any substantial stress in the jet pump body, the converse is not possible. Therefore, failure of a jet pump body cannot occur without the failure of the nozzle riser.

The following factors form the basis for the surveillance requirements:

A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other jet pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will also be used to evaluate jet pump operability.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5%, the flow rates in both recirculation loops will be verified by Control Room monitoring instruments. If the two flow rate values do not differ by more than 15%, riser and nozzle assembly integrity has been verified. If they do differ by 15% or more after correction for the difference in pump speeds, the diffuser to lower plenum differential pressure of all jet pumps will be compared to established jet pump ΔP characteristics. In the event of a failed jet pump nozzle (or riser), the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow. If the jet pump ΔP indications are within 10% of established jet pump ΔP characteristics, jet pump nozzle and riser integrity have been established. If the indicated jet pump ΔP varies from the established jet pump characteristics by more than 10%, indicated core flow will be compared to the core flow derived from loop flow measurements. If the difference between measured and derived core flow rate is 10% or more, a failed jet pump nozzle (or riser) is indicated and the plant shall be shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed.

E. Jet Pumps (Cont)

Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate.

F. Jet Pump Flow Mismatch

The LPCI loop selection logic has been previously described in the Pilgrim Nuclear Power Station FSAR. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. At or below 80% power the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

The flow mismatch restriction also derives from the "Core Flow Coastdown" concern. This concern postulates that if the recirculation loop with the higher flow is broken, the "effective core flow" is determined by the loop with the lower flow. Compared to a matched flow condition, this would start pump coastdown from a lower flow/speed with the reactor power effectively above the rated rod line. Therefore, boiling transition may occur earlier during a postulated LOCA event, which could result in higher calculated peak cladding temperatures (PCTs). Therefore, the purpose of the "Core Flow Coastdown" flow mismatch restriction is to maintain Pilgrim within its analyzed conditions.

Specification 3.6.F allows 30 minutes to correct a mismatch in recirculation pump speeds in order to take manual control of the recirculation pump MG set scoop tube positioner in the event that its control system should fail.

G. Structural Integrity

The Pilgrim Nuclear Power Station Inservice Inspection Program conforms to the requirements of 10CFR50.55a(g). Where practical, the inspection of ASME Section XI Class 1, 2, and 3 components conforms to the edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code required by 10CFR50.55a(g). When implementation of an ASME Code required inspection has been determined to be impractical for PNPS, a request for relief from the inspection requirement is submitted to the NRC in accordance with 10CFR50.55a(g)(5)(iii).

Requests for relief from the ASME Code inspection requirements will be submitted to the NRC prior to the beginning of each 10 year inspection interval for which the inspection requirement is known to be impractical. Requests for relief from inspection requirements which are identified to be impractical during the course of the inspection interval will be reported to the NRC on an annual basis throughout the inspection interval.

Certain ASME Code Class 1, Category B-J pressure retaining welds have been designated as Group I welds. These Group I welds shall be included in the sample of Class I welds requiring inspection during each ten year interval.

I. Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system and all other safety related systems or components be operable during reactor operation.

The visual inspection frequency is based on maintaining a constant level of snubber protection to systems. The cumulative number of inoperable snubbers detected during any inspection interval is the basis for establishment of the subsequent inspection interval and the existing inspection interval should remain in effect until its completion.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable.

Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, and are exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is initiated, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. Initiating this evaluation within 72 hours ensures that prompt corrective action will be afforded.

I. Shock Suppressors (Snubbers) (Cont)

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation. Due to the number and complexity of the relevant interacting factors necessary to develop a comprehensive Service Life Program, this program shall become effective July 1, 1982.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

Suppression Pool

1. At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2 and 3.7.A.3.

   a. Minimum water volume - 84,000 ft^3

   b. Maximum water volume - 94,000 ft^3

   c. Maximum suppression pool bulk temperature during normal continuous power operation shall be $\leq 80^\circ F$, except as specified in 3.7.A.1.e.

   d. Maximum suppression pool bulk temperature during RCIC, HPCI or ADS operation shall be $\leq 90^\circ F$, except as specified in 3.7.A.1.e.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

Suppression Pool

1. a. The suppression chamber water level and temperature shall be checked once per day.

   b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

   c. Whenever there is indication of relief valve operation with the bulk temperature of the suppression pool reaching $160^\circ F$ or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

e. In order to continue reactor power operation, the suppression chamber pool bulk temperature must be reduced to ≤80°F within 24 hours.

f. If the suppression pool bulk temperature exceeds the limits of Specification 3.7.A.1.d, RCIC, HPCI or ADS testing shall be terminated and suppression pool cooling shall be initiated.

g. If the suppression pool bulk temperature during reactor power operation exceeds 110°F, the reactor shall be scrammed.

h. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cool down rates if the pool bulk temperature reaches 120°F.

i. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.17 psid, except as specified in j and k.

j. The differential pressure shall be established within 24 hours of placing the reactor in the run mode following a shutdown. The differential pressure may be reduced to less than 1.17 psid 24 hours prior to a scheduled shutdown.

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

d. Whenever there is indication of relief valve operation with the local temperature of the suppression pool T-quencher reaching 200°F or more, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

e. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

f. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift when the differential pressure is required.

g. Suppression chamber water level shall be recorded at least once each shift when the differential pressure is required.

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

k. The differential pressure may be reduced to less than 1.17 psid for a maximum of four (4) hours for maintenance activities on the differential pressure control system and during required operability testing of the HPCI system, the relief valves, the RCIC system and the drywell-suppression chamber vacuum breakers.

l. If the specifications of Item i, above, cannot be met, and the differential pressure cannot be restored within the subsequent (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty-four (24) hours.

m. Suppression chamber water level shall be maintained between -6 to -3 inches on torus level instrument which corresponds to a downcomer submergence of 3.00 and 3.25 feet respectively.

n. The suppression chamber can be drained if the conditions as specified in Sections 3.5.F.3 and 3.5.F.5 of this Technical Specification are adhered to.

4.7 CONTAINMENT SYSTEMS (Cont)

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

Primary Containment Integrity

2. a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics test at power levels not to exceed 5 Mw(t).

Primary containment integrity means that the drywell and pressure suppression chamber are intact and that all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All blind flanges and manways are closed.

4. All automatic primary containment isolation valves and all instrument line flow check valves are operable except as specified in 3.7.A.2.b.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

Primary Containment Integrity

2. a. The primary containment integrity shall be demonstrated by performing Primary Containment Leak Tests in accordance with 10CFR50 Appendix J, with exemptions as approved by the NRC and exceptions as follows:

1. The main steam line isolation valves shall be tested at a pressure ≥23 psig, and normalized to a value equivalent to 45 psig.

2. Personnel air lock door seals shall be tested at a pressure ≥10 psig. Results shall be normalized to a value equivalent to 45 psig.

3. If the total leakage rates listed below are exceeded, repairs and retests shall be performed to correct the conditions.

1. All double-gasketed seals: 10% L_t (x)

2. All testable penetrations and isolation valves: 60% L_a (x)

3. Any one penetration or isolation valve except main steam line isolation valves: 5% L_t (x)

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

5. All containment isolation check valves are operable or at least one containment isolation valve in each line having an inoperable valve is secured in the isolated position.

Primary Containment Isolation Valves

2. b. In the event any automatic Primary Containment Isolation Valve becomes inoperable, at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition. (This requirement may be satisfied by deactivating the inoperable valve in the isolated condition. Deactivation means to electrically or pneumatically disarm, or otherwise secure the valve.)*

* Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under ORC approved administrative controls.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

4. Any one main steam line isolation valve: 11.5 scf/hr @23 psig.

where x = 45 psig
L_t = .75 L_a
L_a = 1.0% by weight of the contained air @ 45 psig for 24 hrs.

Primary Containment Isolation Valves

2. b. 1 The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

b. Test primary containment isolation valves:

1. Verify power operated primary containment isolation valve operability as specified in 3.13.

2. Verify main steam isolation valve operability as specified in 3.13.

3.7 CONTAINMENT SYSTEMS (Cont)

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

c. At least twice per week the main steam line power operated isolation valves shall be exercised by partial closure and subsequent reopening.

d. Verify reactor coolant system instrument line flow check valve operability as specified in 3.13.

2. b. 2 Whenever a primary containment automatic isolation valve is inoperable, the position of the isolated valve in each line having an inoperable valve shall be recorded daily.

2.c. Continuous Leak Rate Monitor

When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

2.d. Drywell Surfaces

The interior surfaces of the drywell and torus above the water line shall be visually inspected every refueling outage for evidence of deterioration.

LIMITING CONDITION FOR OPERATION

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber - reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber - reactor building breakers shall be 0.5 psig.

b. From and after the date that one of the pressure suppression chamber - reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-pressure suppression chamber vacuum breakers shall be operable except during testing and as stated in Specifications 3.7.A.4.b, c and d, below. Drywell-pressure suppression chamber vacuum breakers shall be considered operable if:

SURVEILLANCE REQUIREMENT

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Verify operability of the pressure suppression chamber-reactor building vacuum breakers as specified in 3.13.

b. Check the associated instrumentation including set points for proper operation every three months.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Periodic Operability Tests

1. Once each month each drywell-pressure suppression chamber vacuum breaker shall be exercised and the operability of the valve and installed position indicators and alarms verified.

2. A drywell to suppression chamber differential pressure decay rate test shall be conducted at least every 3 months.

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

1. The valve is demonstrated to open with the applied force of the installed test actuator as indicated by the position switches and remote position indicating lights.

2. The valve shall return by gravity when released after being opened by remote or manual means, to within 3/32" of the fully closed position.

3. Neither of the two position alarm systems which annunciate on Panel C-7 and Panel 905 when any vacuum breaker opening exceeds 3/32" are in alarm.

b. Any drywell-suppression chamber vacuum breaker may be non-fully closed as determined by the position switches provided that the drywell to suppression chamber differential decay rate is demonstrated to be not greater than 25% of the differential pressure decay rate for the maximum allowable bypass area of $0.2ft^2$.

c. Reactor operation may continue provided that no more than 2 of the drywell-pressure suppression chamber vacuum breakers are determined to be inoperable provided that they are secured or known to be in the closed position.

4.7 CONTAINMENT SYSTEMS (Cont)

A Primary Containment (Cont)

b. During each refueling interval:

1. Each vacuum breaker shall be tested to determine that the disc opens freely to the touch and returns to the closed position by gravity with no indication of binding.

2. Vacuum breaker position switches and installed alarm systems shall be calibrated and functionally tested.

3. At least 25% of the vacuum breakers shall be visually inspected such that all vacuum breakers shall have been inspected following every fourth refueling interval. If deficiencies are found, all vacuum breakers shall be visually inspected and deficiencies corrected.

4. A drywell to suppression chamber leak rate test shall demonstrate that the differential pressure decay rate does not exceed the rate which would occur through a 1 inch orifice without the addition of air or nitrogen.

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

d. If a failure of one of the two installed position alarm systems occurs for one or more vacuum breakers, reactor operation may continue provided that a differential pressure decay rate test is initiated immediately and performed every 15 days thereafter until the failure is corrected. The test shall meet the requirements of Specification 3.7.A.4.b.

5. Oxygen Concentration

a. The primary containment atmosphere shall be reduced to less than 4% oxygen by volume with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.

6. If the specifications of 3.7.A.1 thru 3.7.A.5 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in Cold Shutdown condition within 24 hours.

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

5. Oxygen Concentration

The primary containment oxygen concentration shall be measured and recorded at least twice weekly.

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

7. Containment Atmosphere Dilution

a. Within the 24-hour period after placing the reactor in the Run Mode the Post-LOCA Containment Atmosphere Dilution System must be operable and capable of supplying nitrogen to the containment for atmosphere dilution. If this specification cannot be met, the system must be restored to an operable condition within 30 days or the reactor must be at least in Hot Shutdown within 12 hours.

b. Within the 24-hour period after placing the reactor in the Run Mode, the Nitrogen Storage Tank shall contain a minimum of 1500 gallons of liquid N_2. If this specification cannot be met the minimum volume will be restored within 30 days or the reactor must be in at least Hot Shutdown within 12 hours.

c. There are 2 H_2 analyzers available to serve the drywell.

With only 1 H_2 analyzer operable, reactor operation is allowed for up to 7 days. If the inoperable analyzer is not made fully operable within 7 days, the reactor shall be in a least Hot Shutdown within the next 12 hours.

4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

7. Containment Atmosphere Dilution

a. The post-LOCA containment atmosphere dilution system shall be functionally tested once per operating cycle.

b. The level in the liquid N_2 storage tank shall be recorded weekly.

c. The H_2 analyzers shall be tested for operability once per month and shall be calibrated once per 6 months.

d. Once per month each manual or power operated valve in the CAD system flow path not locked, sealed or otherwise secured in position shall be observed and recorded to be in its correct position.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

With no H_2 analyzer operable, reactor operation is allowed for up to 48 hours. If one of the inoperable analyzers is not made fully operable within 48 hours, the reactor shall be in a least Hot Shutdown within the next 12 hours.

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System

1. Standby Gas Treatment System

a. Except as specified in 3.7.B.1.c below, both trains of the standby gas treatment system and the diesel generators required for operation of such trains shall be operable at all times when secondary containment integrity is required or the reactor shall be shutdown in 36 hours.

b. 1. The results of the in-place cold DOP tests on HEPA filters shall show ≥99% DOP removal. The results of halogenated hydrocarbon tests on charcoal adsorber banks shall show ≥99% halogenated hydrocarbon removal.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS (Cont)

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System

1. Standby Gas Treatment System

a. 1. At least once per operating cycle, it shall be demonstrated that pressure drop across the combined high efficiency filters and charcoal adsorber banks is less than 8 inches of water at 4000 cfm.

2. At least once per operating cycle, demonstrate that the inlet heaters on each train are operable and are capable of an output of at least 14 kW.

3. The tests and analysis of Specification 3.7.B.1.b. shall be performed at least once per operating cycle or following painting, fire or chemical release in any ventilation zone communicating with the system while the system is operating that could contaminate the HEPA filters or charcoal adsorbers.

4. At least once per operating cycle, automatic initiation of each branch of the standby gas treatment system shall be demonstrated, with Specification 3.7.B.1.d satisfied.

3.7 CONTAINMENT SYSTEMS (Cont)

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (Cont)

2. The results of the laboratory carbon sample analysis shall show ≥95% methyl iodide removal at a velocity within 10% of system design, 0.5 to 1.5 mg/m^3 inlet methyl iodide concentration, ≥70% R.H. and ≥190°F. The analysis results are to be verified as acceptable within 31 days after sample removal, or declare that train inoperable and take the actions specified 3.7.B.1.c.

*c From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason, continued reactor operation, irradiated fuel handling, or new fuel handling over spent fuel pool or core is permissible only during the succeeding seven days providing that within 2 hours all active components of the other standby gas treatment train shall be demonstrated to be operable.

4.7 CONTAINMENT SYSTEMS (Cont)

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (Cont)

5. Each train of the standby gas treatment system shall be operated for at least 15 minutes per month.

6. The tests and analysis of Specification 3.7.B.1.b.2 shall be performed after every 720 hours of system operation.

b. 1. In-place cold DOP testing shall be performed on the HEPA filters after each completed or partial replacement of the HEPA filter bank and after any structural maintenance on the HEPA filter system housing which could affect the HEPA filter bank bypass leakage.

2. Halogenated hydrocarbon testing shall be performed on the charcoal adsorber bank after each partial or complete replacement of the charcoal adsorber bank or after any structural maintenance on the charcoal adsorber housing which could affect the charcoal adsorber bank bypass leakage.

* During RFO #9, one train can be without its safety-related bus and/or emergency diesel generator without entering the LCO action statement provided the following conditions are met:

- Fuel movement will not occur until five days following reactor shutdown.
- Prior to and during fuel movement, the SBO D/G or the Shutdown Transformer is required to be operable and capable of supply power to the emergency bus.
- Fuel movement will not occur until the reactor vessel is flooded up to elevation 114'.
- The train of SGTS and CRHEAF without its safety related bus or without its emergency diesel generator will have power supplied from a normal offsite source via a non safety-related bus. The normal offsite sources consist of either the Startup Transformer on Unit Auxiliary Transformer (Backscuttle Mode).

3.7 CONTAINMENT SYSTEMS (Cont)

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (Cont)

d. Fans shall operate within ± 10% of 4000 cfm.

*e Except as specified in 3.7.B.1.c, both trains of the Standby Gas Treatment System shall be operable during irradiated fuel handling, or new fuel handling over the spent fuel pool or core. If the system is not operable, fuel movement shall not be started. Any fuel assembly movement in progress may be completed.

2. Control Room High Efficiency Air Filtration System

*a Except as specified in Specification 3.7.B.2.c below, both trains of the Control Room High Efficiency Air Filtration System used for the processing of inlet air to the control room under accident conditions and the diesel generator(s) required for operation of each train of the system shall be operable whenever secondary containment integrity is required and during fuel handling operations.

4.7 CONTAINMENT SYSTEMS (Cont)

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (Cont)

2. Control Room High Efficiency Air Filtration System

a. At least once per operating cycle the pressure drop across each combined filter train shall be demonstrated to be less than 6 inches of water at 1000 cfm or the calculated equivalent.

* During RFO #9, one train can be without its safety-related bus and/or its emergency diesel generator without entering the LCO action statement provided the conditions listed on page 3/4.7-12 are met.

3.7 CONTAINMENT SYSTEMS (Cont)

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (Cont)

b. 1. The results of the in-place cold DOP tests on HEPA filters shall show ≥99% DOP removal. The results of the halogenated hydrocarbon tests on charcoal adsorber banks shall show ≥99% halogenated hydrocarbon removal when test results are extrapolated to the initiation of the test.

2. The results of the laboratory carbon sample analysis shall show ≥95% methyl iodide removal at a velocity within 10% of system design, 0.05 to 0.15 mg/m^3 inlet methyl iodide concentration, ≥70% R.H., and ≥125°F. The analysis results are to be verified as acceptable within 31 days after sample removal, or declare that train inoperable and take the actions specified in 3.7.B.2.c.

4.7 CONTAINMENT SYSTEMS (Cont)

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (Cont)

b. 1. The tests and analysis of Specification 3.7.B.2.b shall be performed once per operating cycle or following painting, fire or chemical release in any ventilation zone communicating with the system while the system is operating.

2. In-place cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing which could affect the HEPA filter bank bypass leakage.

3. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing which could affect the charcoal adsorber bank bypass leakage.

4. Each train shall be operated with the heaters in automatic for at least 15 minutes every month.

5. The test and analysis of Specification 3.7.B.2.b.2 shall be performed after every 720 hours of system operation.

3.7 CONTAINMENT SYSTEMS (Cont)

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (Cont)

*c From and after the date that one train of the Control Room High Efficiency Air Filtration System is made or found to be incapable of supplying filtered air to the control room for any reason, reactor operation or refueling operations are permissible only during the succeeding 7 days providing that within 2 hours all active components of the other CRHEAF train shall be demonstrated operable. If the system is not made fully operable within 7 days, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within the next 36 hours and irradiated fuel handling operations shall be terminated within 2 hours. Fuel handling operations in progress may be completed.

d. Fans shall operate within ± 10% of 1000 cfm.

4.7 CONTAINMENT SYSTEMS (Cont)

B. Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (Cont)

c. At least once per operating cycle demonstrate that the inlet heaters on each train are operable and capable of an output of at least 14 kw.

d. Perform an instrument functional test on the humidistats controlling the heaters once per operating cycle.

* During RFO #9, one train can be without its safety-related bus and/or its emergency diesel generator without entering the LCO action statement provided the conditions listed on page 3/4.7-12 are met.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS (Cont)

C. Secondary Containment

1 Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.

a. The reactor is subcritical and Specification 3.3.A is met.

b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

d. The fuel cask or irradiated fuel is not being moved in the reactor building.

2. If Specification 3.7.C.1 cannot be met, procedures shall be initiated to establish conditions listed in Specification 3.7.C.1.a through d.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS (Cont)

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

a. A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. Such tests shall demonstrate the capability to maintain 1/4 inch of water vacuum under calm wind (<5 mph) conditions with a filter train flow rate of not more than 4000 cfm.

b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

c. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (5 mph) conditions with a filter train flow rate of not more than 4000 cfm, shall be demonstrated at each refueling outage prior to refueling.

BASES:

3/4.7 CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination limit the off-site doses to values less than those suggested in 10CFR100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception was made to this requirement during initial core loading and while the low power test program was being conducted and ready access to the reactor vessel was required. There was no pressure on the system at this time, thus greatly reducing the chances of a pipe break. Should this type of testing be necessary in the future, the reactor may be taken critical; however, restrictive operating procedures would be in effect again to minimize the probability of an accident. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the secondary containment and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10CFR100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the maximum of 62 psig. Maximum water volume of 94,000 ft^3 results in a downcomer submergency of 4'-0" and the minimum volume of 84,000 ft^3 results in a submergence approximately 12-inches less. Mark I Containment Long Term Program Quarter Scale Test Facility (QSTF) testing at a downcomer submergency of 3.25 feet and 1.17 psi wetwell to drywell pressure differential shows a significant suppression chamber load reduction and Long Term Program analysis and modifications are based on the above submergence and differential pressure.

Should it be necessary to drain the suppression chamber, provision will be made to maintain those requirements as described in Section 3.5.F BASES of this Technical Specification.

A. Primary Containment (Cont)

Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the pressure suppression pool is maintained below 200°F during any period of relief-valve operation with sonic conditions at the discharge exit. Analysis has been performed to verify that the local pool temperature will stay below 200°F and the bulk temperature will stay below 160°F for all SRV transients. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

A. Primary Containment (Cont)

Primary Containment Testing

The primary containment pre-operational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The calculated peak drywell pressure is about 45 psig which would rapidly reduce to 27 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5%/day at a pressure of 56 psig. Based on the calculated containment pressure response discussed above, the primary containment pre-operational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at t e primary containment maximum allowable accident leak rate of 1.25%/day at 45 psig. Calculations made by the AEC staff with this leak rate and a standby gas treatment system filter efficiency of 95% for halogens and assuming the fission product release fractions stated in TID 14844, show that the maximum total whole body passing cloud dose is about 13 REM and the maximum total thyroid dose is about 110 REM at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population zone distance of 4.3 miles are about 3 REM total whole body and 70 REM total thyroid. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site dose and 10CFR100 guidelines.

The maximum allowable test leak rate is 1.0%/day at a pressure of 45 psig. This value for the test condition was derived from the maximum allowable accident leak rate of 1.25%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air whereas under test conditions the test medium would be air at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 as determined from the guide on containment testing.

Establishing the test limit of 1.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness

A. Primary Containment (Cont)

capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate or the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is in accordance with 10CFR50 App. J.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

Group 1 - process lines are isolated by reactor vessel low-low water level in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, main steam space high temperature, or reactor vessel high water level.

Group 2 - isolation valves are closed by reactor vessel low water level or high drywell pressure. The group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the group 2 isolation signal by a transient or spurious signal.

Group 3 - isolation valves can only be opened when the reactor is at low pressure and the core standby cooling systems are not required. Also, since the reactor vessel could potentially be drained through these process lines, these valves are closed by low water level.

A. Primary Containment (Cont)

Group 4 and 5 - process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of group 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low level isolation is provided.

Group 7 - The HPCI vacuum breaker line is designed to remain operable when the HPCI system is required. The signals which initiate isolation of the HPCI vacuum breaker line are indicative of a break inside containment and reactor pressure below that at which HPCI can operate.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance, prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

A. Primary Containment (Cont)

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment. A program for periodic testing and examination of the excess flow check valves is in place.

Primary Containment Painting

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, assures the paint is intact. Experience at Pilgrim Station and other BWR's with this type of paint indicates that the inspection interval is adequate.

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure. One valve may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the 10 drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling to the design limit of 2 psig. They are sized on the basis of the Bodega Bay pressure suppression system tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows a 5 psig vacuum; therefore, with two vacuum relief valves secured in the closed position and eight operable valves, containment integrity is not impaired.

Reactor operation is permissible if the bypass area between the primary containment drywell and suppression chamber does not exceed an allowable area. The allowable bypass area is based upon analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is 0.2 ft^2, which is equivalent to all vacuum breakers open 3/32". (See letters from Boston Edison to the Directorate of Licensing, dated May 15, 1973 and October 22, 1974)

Reactor operation is not permitted if differential pressure decay rate is demonstrated to exceed 25% of allowable, thus providing a margin of safety for the primary containment in the event of a small break in the primary system.

A. Primary Containment (Cont)

Each drywell suppression chamber vacuum breaker is equipped with three switches. One switch provides full open indication only. Another switch provides closed indication and an alarm on Panel C-7 should any vacuum breaker come off its closed seat by greater than 3/32". The third switch provides a separate and redundant alarm on Panel 905 should any vacuum breaker come off its closed seat by greater than 3/32". The two alarms above are those referred to in Section 3.7.A.4.a.3 and 3.7.A.4.d.

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The 4% oxygen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance. Mark I Containment Long Term Program testing showed that maintaining a drywell to

wetwell pressure differential to keep the suppression chamber downcomer legs clear of water significantly reduced suppression chamber post LOCA hydrodynamic loads. A pressure of 1.17 psid is required to sufficiently clear the water legs of the downcomers without bubbling nitrogen into the suppression chamber at the 3.00 ft. downcomer submergence which corresponds to approximately 84,000 ft.^{3} of water. Maximum downcomer submergence is 3.25 ft. at operating suppression chamber water level. The above pressure differential and submergence number are used in the Pilgrim I Plant Unique Analysis.

Post LOCA Atmosphere Dilution

In order to ensure that the containment atmosphere remains inerted, i.e. the oxygen-hydrogen mixture below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. A minimum of 1500 gallons of liquid N_2 in the storage tank assures that a three-day supply of N_2 for post-LOCA containment inerting is available. Since the inerting makeup system is continually functioning, no periodic testing of the system is required.

The Post-LOCA Containment Atmospheric Dilution (CAD) System is designed to meet the requirements of AEC Regulatory Guides 1.3, 1.7 and 1.29, ASME Section III, Class 2 (except for code stamping) and seismic Class I as defined in the PNPS FSAR.

In summary, the limiting criteria are:

1. Maintain hydrogen concentration in the containment during post-LOCA conditions to less than 4%.

2. Limit the buildup in the containment pressure due to nitrogen addition to less than 28 psig.

3. To limit the offsite dose due to containment venting (for pressure control) to less than 300 Rem to the thyroid.

By maintaining at least a 3-day supply of N_2 on site there will be sufficient time after the occurrence of a LOCA for obtaining additional nitrogen supply from local commercial sources.(1) The system design contains sufficient redundancy to ensure its reliability. Thus, it is sufficient to test the operability of the whole system once per operating cycle. The H_2 analyzers will provide redundancy for the drywell i.e., there are two H_2 analyzers for the Unit. By permitting reactor operation for 7 days with one of the two H_2 analyzers inoperable, redundancy of analyzing capability will be maintained while not imposing an immediate interruption in plant operation. Monthly

(1) As listed in Pilgrim Nuclear Power Station Procedure No. 5.4.6 "Post Accident Venting".

testing of the analyzers using H_2 will be adequate to ensure the system's readiness because of the design. Since the analyzers are normally not in operation there will be little deterioration due to use. In order to determine H_2 concentration, the analyzers must be warmed up 6 hours prior to putting into service. This time frame is acceptable for accident conditions because a 4% H_2 level will not be reached in the drywell until 16 hours following the accident. Due to nitrogen addition, the pressure in the containment after a LOCA will increase with time. Under the worst expected conditions the containment pressure will reach 28 psig in approximately 45 days. If and when that pressure is reached, venting from the containment shall be manually initiated per the requirements of 10CFR50.44. The venting path will be through the Standby Gas Treatment system in order to minimize the off site dose.

B.1 Standby Gas Treatment System

The Standby Gas Treatment System is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Upon containment isolation, both standby gas treatment fans are designed to start to bring the reactor building pressure negative so that all leakage should be in leakage. After a preset time delay, the standby fan automatically shuts down so the reactor building pressure is maintained approximately 1/4 inch of water negative. Should one system fail to start, the redundant system is designed to start automatically. Each of the two trains has 100% capacity.

High Efficiency Particulate Air (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA filter efficiency of at least 99 percent removal of cold DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of at least 95 percent for expected accident conditions. The specified efficiencies for the charcoal and particulate filters is sufficient to preclude exceeding 10CFR100 guidelines for the accidents analyzed. The analysis of the loss of coolant accident assumed a charcoal adsorber efficiency of 95% and TID 14844 fission product source terms, hence, installing two banks of adsorbers and filters in each train provides adequate margin. A 14 kW heater maintains relative humidity below 70% in order to ensure the efficient removal of methyl iodide on the impregnated charcoal adsorbers. Considering the relative simplicity of the heating circuit, the test frequency of once/operating cycle is adequate to demonstrate operability.

Air flow through the filters and charcoal adsorbers for 15 minutes each month assures operability of the system. Since the system heaters are automatically controlled, the air flowing through the filters and adsorbers will be ≤70% relative humidity and will have the desired drying effect.

Tests of impregnated charcoal identical to that used in the filters indicate that a shelf life of five years leads to only minor decreases in methyl iodide removal efficiency. Hence, the frequency of laboratory carbon sample analysis is adequate to demonstrate acceptability. Since adsorbers must be removed to perform this analysis this frequency also minimizes the system out of service time as a result of surveillance testing. In addition, although the halogenated hydrocarbon testing is basically a leak test, the adsorbers have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodide, the testing also gives an indication of the relative efficiency of the installed system. The 31 day requirement for the ascertaining of test results ensures that the ability of the charcoal to perform its designed function is demonstrated and known in a timely manner.

The required Standby Gas Treatment System flow rate is that flow, less than or equal to 4000 CFM which is needed to maintain the Reactor Building at a 0.25 inch of water negative pressure under calm wind conditions. This capability is adequately demonstrated during Secondary Containment Leak Rate Testing performed pursuant to Technical Specification 4.7.C.1.c.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters or adsorbers, thus reducing their reserve capacity too quickly. The filter testing is performed pursuant to appropriate procedures reviewed and approved by the Operations Review Committee pursuant to Section 6 of these Technical Specifications. The in-place testing of charcoal filters is performed by injecting a halogenated hydrocarbon into the system upstream of the charcoal adsorbers. Measurements of the concentration upstream and downstream are made. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. A similar procedure substituting dioctyl phthalate for halogenated hydrocarbon is used to test the HEPA filters.

Pressure drop tests across filter and adsorber banks are performed to detect plugging or leak paths though the filter or adsorber media. Considering the relatively short times the fans will be run for test purposes, plugging is unlikely and the test interval of once per operating cycle is reasonable.

System drains and housing gasket doors are designed such that any leakage would be inleakage from the Standby Gas Treatment System Room. This ensures that there will be no bypass of process air around the filters or adsorbers.

Only one of the two Standby Gas Treatment Systems (SBGTS) is needed to maintain the secondary containment at a 0.25 inch of water negative pressure upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling activities may continue while repairs are being made. In the event one SBGTS is inoperable, the redundant system's active components will be tested within 2 hours. This substantiates the availability of the operable system and justifies continued reactor or refueling operations.

If both trains of SBGTS are inoperable, the plant is brought to a condition where the SBGTS is not required.

B.2 Control Room High Efficiency Air Filtration System

The Control Room High Efficiency Air Filtration System is designed to filter intake air for the control room atmosphere during conditions when normal intake air may be contaminated. Following manual initiation, the Control Room High Efficiency Air Filtration System is designed to position dampers and start fans which divert the normal air flow through charcoal adsorbers before it reaches the control room.

High Efficiency Particulate Air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. A second bank of HEPA filters is installed downstream of the charcoal filter.

The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of cold DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of at least 90 percent for expected accident conditions. Tests of impregnated charcoal identical to that used in the filters indicate that a shelf life of five years leads to only minor decreases in methyl iodine removal efficiency. Hence, the frequency of laboratory carbon sample analysis is adequate to demonstrate acceptability. Since adsorbers must be removed to perform this analysis, this frequency also minimizes the system out of service time as a result of surveillance testing. In addition, although the halogenated hydrocarbon testing is basically a leak test, the adsorbers have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodide, the testing also gives an indication of the relative efficiency of the installed system. The 31 day requirement for the ascertaining of test results ensures that the ability of the charcoal to perform its designed function is demonstrated and known in a timely manner.

Determination of the system pressure drop once per operating cycle provides indication that the HEPA filters and charcoal adsorbers are not clogged by excessive amounts of foreign matter and that no bypass routes through the filters or adsorbers had developed. Considering the relatively short times the systems will be operated for test purposes, plugging is unlikely and the test interval of once per operating cycle is reasonable.

B.2 Control Room High Efficiency Air Filtration System (Cont)

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters or adsorbers, thus reducing their reserve capacity too quickly. The filter testing is performed pursuant to appropriate procedures reviewed and approved by the Operations Review Committee pursuant to Section 6 of these Technical Specifications. The in-place testing of charcoal filters is performed by injecting a halogenated hydrocarbon into the system upstream of the charcoal adsorbers. Measurements of the concentration upstream and downstream are made. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. A similar procedure substituting dioctyl phthalate for halogenated hydrocarbon is used to test the HEPA filters.

Air flow through the filters and charcoal adsorbers for 15 minutes each month assures operability of the system. Since the system heaters are automatically controlled, the air flowing through the filters and adsorbers will be ≤ 70% relative humidity and will have the desired drying effect.

If one train of the system is found to be inoperable, there is no immediate threat to the control room, and reactor operation or fuel handling may continue for a limited period of time while repairs are being made. In the event one CRHEAF train is inoperable, the redundant system's active components will be tested within 2 hours. If both trains of the CRHEAF system are inoperable, the reactor will be brought to a condition where the Control Room High Efficiency Air Filtration System is not required.

C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water negative pressure within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

LIMITING CONDITIONS FOR OPERATION

3.8 RADIOACTIVE EFFLUENTS

A. Liquid Effluents Concentration

Applicability:

At all times.

Specification:

1. The concentration of radioactive material released at any time from the site to areas at and beyond the site boundary shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration of individual isotopes shall be limited to 2 x 10^{-4} μ Ci/ml.

Action

With the concentration of radioactive material released from the site to areas at and beyond the site boundary exceeding the above limits, without delay restore concentration within the above limits.

B. Radioactive Liquid Effluent Instrumentation

Applicability:

As shown in Table 3.8-1.

Specification:

1. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.8-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.8.A.1 are not exceeded

SURVEILLANCE REQUIREMENTS

4.8 RADIOACTIVE EFFLUENTS

A. Liquid Effluents Concentration

Specification:

1. The radioactivity content of each batch of radioactive liquid waste to be discharged shall be determined prior to release by sampling and analysis in accordance with Table 4.8-1.

2. The results of pre-release analyses shall be used with calculational methods in the Offsite Dose Calculation Manual (ODCM) to assure that the concentration at the point of release is limited to the values in Specification 3.8.A.1.

B. Radioactive Liquid Effluent Instrumentation

Specification:

1. The setpoints for monitoring instrumentation shall be determined in accordance with the ODCM.

2. Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated operable at the frequencies shown in Table 4.8-2.

3.8 RADIOACTIVE EFFLUENTS (Cont)

B. Radioactive Liquid Effluent Instrumentation (Cont)

during periods when liquid wastes are being discharged via the radwaste discharge header.

For releases other than the radwaste discharge header, the above specification does not apply, these releases shall be made in accordance with Action 1 of Table 3.8-1.

Action

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Specification 3.8.A.1 are met, without delay suspend the release of radioactive liquid effluents monitored by the affected channel or change the setpoint so that it is acceptably conservative or declare the channel inoperable.

b. With one or more radioactive liquid effluent monitoring instrumentation channels inoperable, take the action shown in Table 3.8-1.

C. Liquid Radwaste Treatment

Applicability:

At all times.

Specification:

1. The liquid radwaste treatment system shall be maintained and used to reduce the radioactive materials in liquid wastes

4.8 CONTAINMENT SYSTEMS (Cont)

C. Liquid Radwaste Treatment

Specification:

1. Doses due to liquid releases at and beyond the site boundary shall be calculated at least once per 31-day period in accordance with the ODCM, only if releases in that period have occurred.

3.8 RADIOACTIVE EFFLUENTS (Cont)

C. Liquid Radwaste Treatment (Cont)

prior to their discharge when the dose due to liquid effluent releases to areas at and beyond the site boundary averaged over a 31-day period would exceed 0.06 mrem to the total body or 0.20 mrem to any organ.

Action

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days a special report which includes the following information:

1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability.

2. Action(s) taken to restore the inoperable equipment to operable status.

3. Summary description of action(s) taken to prevent a recurrence.

D. Gaseous Effluents Dose Rate

Applicability:

At all times.

Specification:

1. The instantaneous dose rate due to radioactive materials released in gaseous effluents

4.8 RADIOACTIVE EFFLUENTS (Cont)

C. Liquid Radwaste Treatment (Cont)

2. The liquid radwaste treatment system schematic is shown in Figure 4.8-1.

D. Gaseous Effluents Dose Rate

Specification:

1. The instantaneous dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.8.D.1.a on a continuous basis using the noble gas activity monitors

3.8 RADIOACTIVE EFFLUENTS (Cont)

D. Gaseous Effluents Dose Rate (Cont)

from the site to areas at and beyond the site boundary (see FSAR Figure 1.6-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

Action

With the instantaneous dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

E. Radioactive Gaseous Effluent Instrumentation

Applicability:

As shown in Table 3.8-2.

Specification:

1. The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.8-2 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.8.D.1 are not exceeded.

4.8 RADIOACTIVE EFFLUENTS (Cont)

D. Gaseous Effluents Dose Rate (Cont)

with appropriate setpoints and in accordance with the ODCM.

2. The instantaneous dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.8.D.1.b in accordance with the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.8-3.

E. Radioactive Gaseous Effluent Instrumentation

Specification:

1. The setpoints shall be determined in accordance with ODCM.

2. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated operable at the frequencies shown in Table 4.8-4.

3.8 RADIOACTIVE EFFLUENTS (Cont)

E. Radioactive Gaseous Effluent Instrumentation (Cont)

Action

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Specification 3.8.D.1 are met, change the setpoint so that it is acceptably conservative or declare the channel inoperable.

b. With one or more radioactive gaseous effluent monitoring instrumentation channels inoperable, take the action shown in Table 3.8-2.

F. Gaseous Effluent Treatment

Applicability:

The augmented offgas system shall be put into service prior to reaching 50 percent reactor power during startup.

Action

a. With gaseous effluents being discharged for more than 14 days without treatment, prepare and submit to the Commission within 30 days, a special report which includes the following information:

1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability.

2. Action(s) taken to restore the inoperable equipment to operable status.

4.8 RADIOACTIVE EFFLUENTS (Cont)

F. Gaseous Effluent Treatment

Specification:

1. Augmented offgas annunciator operability shall be verified once per 12 hours.

2. The concentration of hydrogen in the augmented offgas treatment system shall be determined to be within the limits of Specification 3.8.F.1 by continuously monitoring the waste gases in the augmented offgas treatment system with the hydrogen monitor which is required to be operable by Table 3.8-2.

3.8 RADIOACTIVE EFFLUENTS (Cont)

F. Gaseous Effluent Treatment (Cont)

3. Summary description of action(s) taken to prevent a recurrence.

Specification:

1. The concentration of hydrogen in the augmented offgas treatment system shall be limited to less than or equal to 2 percent by volume at the outlet of the augmented offgas recombiner. See also Action 5 for Item 4.a on Table 3.8-2.

Action

a. With the concentration of hydrogen in the augmented offgas treatment system greater than 2 percent by volume but less than or equal to 4 percent by volume, restore the concentration of hydrogen to within the limit within 48 hours or be in a cold shutdown condition within 24 hours.

G. Main Condenser

Applicability:

At all times when steam is available to the air ejectors.

Specification:

1. The gross radioactivity (beta and/or gamma) release rate of noble gases measured at the steam jet air ejector shall be limited to 500,000 μ Ci/sec (referenced to a 30-minute holdup).

4.8 RADIOACTIVE EFFLUENTS (Cont)

G. Main Condenser

Specification:

1. The gross radioactivity (beta and/or gamma) release rate of noble gases from the steam jet air ejector shall be determined to be within the limit of Specification 3.8.G.1 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the discharge of the steam jet air ejector (prior to dilution and/or discharge):

a. At least once per 31 days.

LIMITING CONDITIONS FOR OPERATION

3.8 RADIOACTIVE EFFLUENTS (Cont)

G. Main Condenser (Cont)

Action

With the gross radioactivity (beta and/or gamma) release rate of noble gases at the steam jet air ejector exceeding 500,000 μ Ci/sec (referenced to a 30-minute holdup), restore the gross radioactivity release rate to within the limit within 72 hours or be in at least hot standby within the next 12 hours. See also Action 1 for Item 3.a on Table 3.8-2.

H. Mechanical Vacuum Pump

Specification:

1. The mechanical vacuum pump shall be capable of being isolated and secured on a signal of high radioactivity in the steam lines whenever the main steam isolation valves are open.

2. If the limits of Specification 3.8.H.1 are not met, the vacuum pump shall be isolated.

SURVEILLANCE REQUIREMENTS

4.8 RADIOACTIVE EFFLUENTS (Cont)

G. Main Condenser (Cont)

b. When the average daily gross radioactivity release rate increases by 50 percent over the previous day, after factoring out increases due to changes in reactor thermal power level.

H. Mechanical Vacuum Pump

Specification:

1. At least once during each operating cycle verify automatic securing and isolation of the mechanical vacuum pump.

TABLE 3.8-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Applicability	Action[2]
1. Gross Radioactivity Monitors Providing Automatic Termination of Release			
a. Liquid Radwaste Effluent Line	1	During actual discharge of liquid wastes	1
2. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line	1	During actual discharge of liquid wastes	2
b. Discharge Canal[1]	NA	During actual discharge of liquid wastes	3

1 Flow will be estimated based on the design flow rate of the operating circulating water pumps and/or the operating salt service water pumps.

2 ACTION 1 With the number of operable channels less than required by the minimum channels operable requirement, effluent releases may be resumed provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.8.A.1, and

b. An independent verification of the release rate calculations is performed, and

c. An independent verification of the discharge valving is performed.

ACTION 2 With the number of operable channels less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that the flow rate is verified at least once per 4 hours during actual releases.

ACTION 3 Suspend all radioactive liquid effluent discharges if no dilution water is available.

TABLE 3.8-2

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Applicability	Parameter	Action[a]
1. Main Stack Effluent Monitoring System				
a. Noble Gas Activity Monitor - Providing Alarm	1	1	Radioactivity Rate Measurement	3
b. Iodine Sampler Cartridge	1	1	Collect Halogen Sample	4
c. Particulate Sampler Filter	1	1	Collect Particulate Sample	4
d. Effluent System Flow Rate Measuring Device	1	1	System Flow Rate Measurement	2
e. Sampler Flow Rate Measuring Device	1	1	Sampler Flow Rate Measurement	2
2. Reactor Building Ventilation Effluent Monitoring System				
a. Noble Gas Activity Monitor - Providing Alarm	1	1	Radioactivity Rate Measurement	3
b. Iodine Sampler Cartridge	1	1	Collect Halogen Sample	4
c. Particulate Sampler Filter	1	1	Collect Particulate Sample	4

TABLE 3.8-2 (Cont)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Applicability	Parameter	Action[4]
2. Reactor Building Ventilation Effluent Monitoring System (Continued)				
d. Effluent System Flow Rate Measurement Device	1	1	System Flow Rate Measurement	2
e. Sampler Flow Rate Measurement Device	1	1	Sampler Flow Rate Measurement	2
3. Steam Jet Air Ejector Radioactivity Monitor				
a. Noble Gas Activity Monitor (Providing alarm and auto-isolation of stack)	1	3	Noble Gas Radioactivity rate Measurement	1
4. Augmented Offgas Treatment System Explosive Gas Monitoring				
a. Hydrogen Monitor	1	2	Hydrogen Concentration Measurement	5

TABLE 3.8-2 (Cont)

TABLE NOTATION

[1] During releases via this pathway.

[2] During augmented offgas treatment system operation.

[3] During operation of the steam jet air ejector.

[4] ACTION 1 With the number of operable channels less than required by the minimum channels operable requirement, gases from the steam jet air ejector may be released to the offgas system for up to 72 hours provided:

a. The augmented offgas treatment system is not bypassed, and

b. The offgas holdup system noble gas activity effluent monitor (downstream) is operable.

Otherwise, be in at least hot standby within 12 hours.

ACTION 2* With the number of operable channels less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 3* With the number of operable channels less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for activity within 24 hours.

ACTION 4* With the number of operable channels less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.8-3.

ACTION 5 With the number of operable channels less than required by the minimum channels operable requirement, operation of the augmented offgas holdup system may continue provided grab samples are collected at least once per 24 hours, analyzed within the following 4 hours, and the proper function of the recombiner is assured by monitoring recombiner temperature.

*Note: (For Actions 2, 3, and 4) If the instruments are not returned to operable status within 30 days, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

TABLE 4.8-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[c]	Each Batch	Prior to Release Each Batch	Principal Gamma Emitters[d]	5×10^{-7}
1. Non-treatable Releases (e.g., Neutralizer Sumps) and 2. Treatable Releases (e.g., Radwaste Tanks)			I-131	1×10^{-6}
			Dissolved and Entrained Gases	1×10^{-5}
	Composite for Each Batch	Monthly Composite[b]	H-3	1×10^{-5}
			Gross alpha	1×10^{-7}
	Composite from Each Batch	Quarterly Composite[b]	Sr-89, Sr-90	5×10^{-8}
			FE-55	1×10^{-6}
B. Continuous Releases				
1. Salt Service Water	Weekly grab sample	Weekly	Principal Gamma Emitters	5×10^{-7}

TABLE 4.8-1 (Cont)

TABLE NOTATION

(a) Refer to ODCM for LLD definition.

(b) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

(c) A batch release is the discharge of liquid wastes of a discrete volume.

(d) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall be analyzed and reported in the Semiannual Radioactive Effluent Release Report.

TABLE 4.8-2

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Channel Calibration	Channel Functional Test
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation				
a. Liquid Radwaste Effluents Line	1	NA	Once per 18 months	Quarterly
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	1	NA	Once per 18 months	Quarterly

[1]During or prior to release via this pathway.

[2]Previously established calibration procedures will be used for these requirements.

TABLE 4.8-3

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	(LLD)[a] (μCi/ml)
Main Stack and Rx Bldg. Vent	Monthly Grab Sample	Monthly	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[d]	Weekly Charcoal[c] Sample	I-131	1×10^{-12}
	Continuous[d]	Weekly Particulate[c] Sample	Principal Gamma Emitters[b] (I-131, others)	1×10^{-11}
	Continuous[d]	Monthly Composite Particulate Sample	gross alpha	1×10^{-11}
	Continuous[d]	Quartely Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[d]	Continuous Noble Gas Monitor	Noble Gases Gross Gamma	1×10^{-6}

TABLE 4.8-3 (Cont)

TABLE NOTATION

(a) Refer to ODCM for LLD definition.

(b) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions; and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall be analyzed and reported in the Semiannual Radioactive Effluent Release Report.

(c) When the average daily gross radioactivity release rate increases by 50 percent over the previous day (after factoring out power level changes), the iodine and particulate filters shall be analyzed to determine the release rate for iodines and particulates.

(d) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specification 3.8.D.

TABLE 4.8-4

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
1. Main Stack Effluent Monitoring System				
a. Noble Gas Activity Monitor (Two channels)	Daily[1]	Monthly	Once per 18 months[4]	Quarterly
b. Iodine Sampler Cartridge	NA	NA	NA	NA
c. Particulate Sampler Filter	NA	NA	NA	NA
d. Effluent System Flow Rate Measuring Device	Daily[1]	NA	Once per 18 months	Quarterly
e. Sampler Flow Rate Measuring Device	Daily[1]	NA	Once per 18 months	Quarterly
2. Reactor Building Ventilation Effluent Monitoring System				
a. Noble Gas Activity Monitor	Daily[1]	Monthly	Once per 18 months[4]	Quarterly
b. Iodine Sampler Cartridge	NA	NA	NA	NA
c. Particulate Sampler Filter	NA	NA	NA	NA
d. Effluent System Flow Rate Measuring Device	Daily[1]	NA	Once per 18 months	Quarterly

TABLE 4.8-4 (Cont)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
e. Sampler Flow Rate Measuring Device	Daily[1]	NA	Once per 18 months cycle	Quarterly
3. Steam Jet Air Ejector Radioactivity Monitor				
a. Noble Gas Activity Monitor	Daily[3]	NA	Once per operating cycle[4]	Quarterly
4. Augmented Offgas Treatment System Explosive Gas Monitoring System				
a. Hydrogen Monitor	Daily[2]	NA	Quarterly[5]	Monthly

[1]During releases via this pathway.
[2]During augmented offgas treatment system operation.
[3]During operation of the steam jet air ejector.
[4]Previously established calibration procedures will be used for these requirements.
[5]Calibrate at 2 points with standard gas samples differing by at least 1% but not exceeding 4%

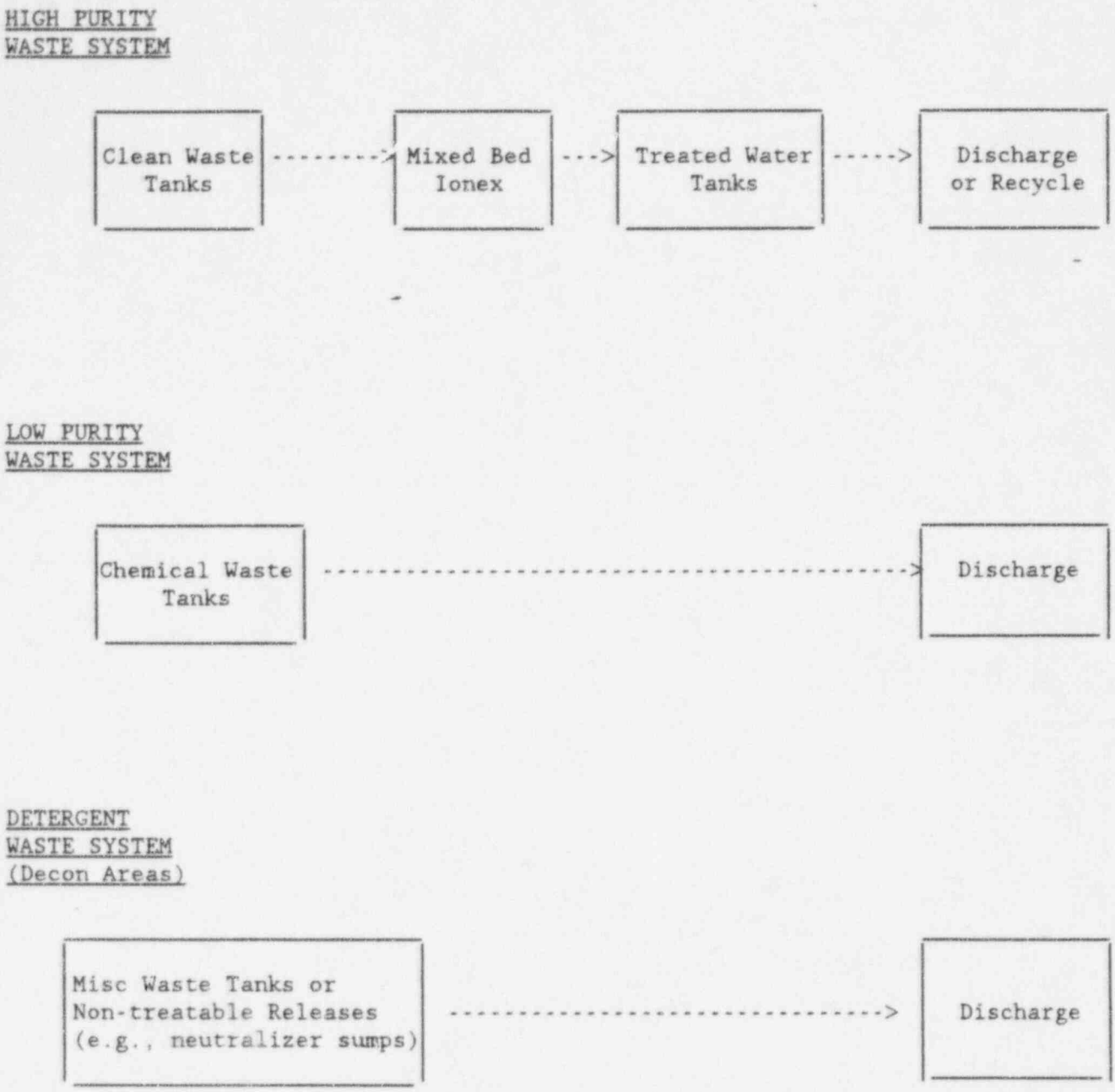
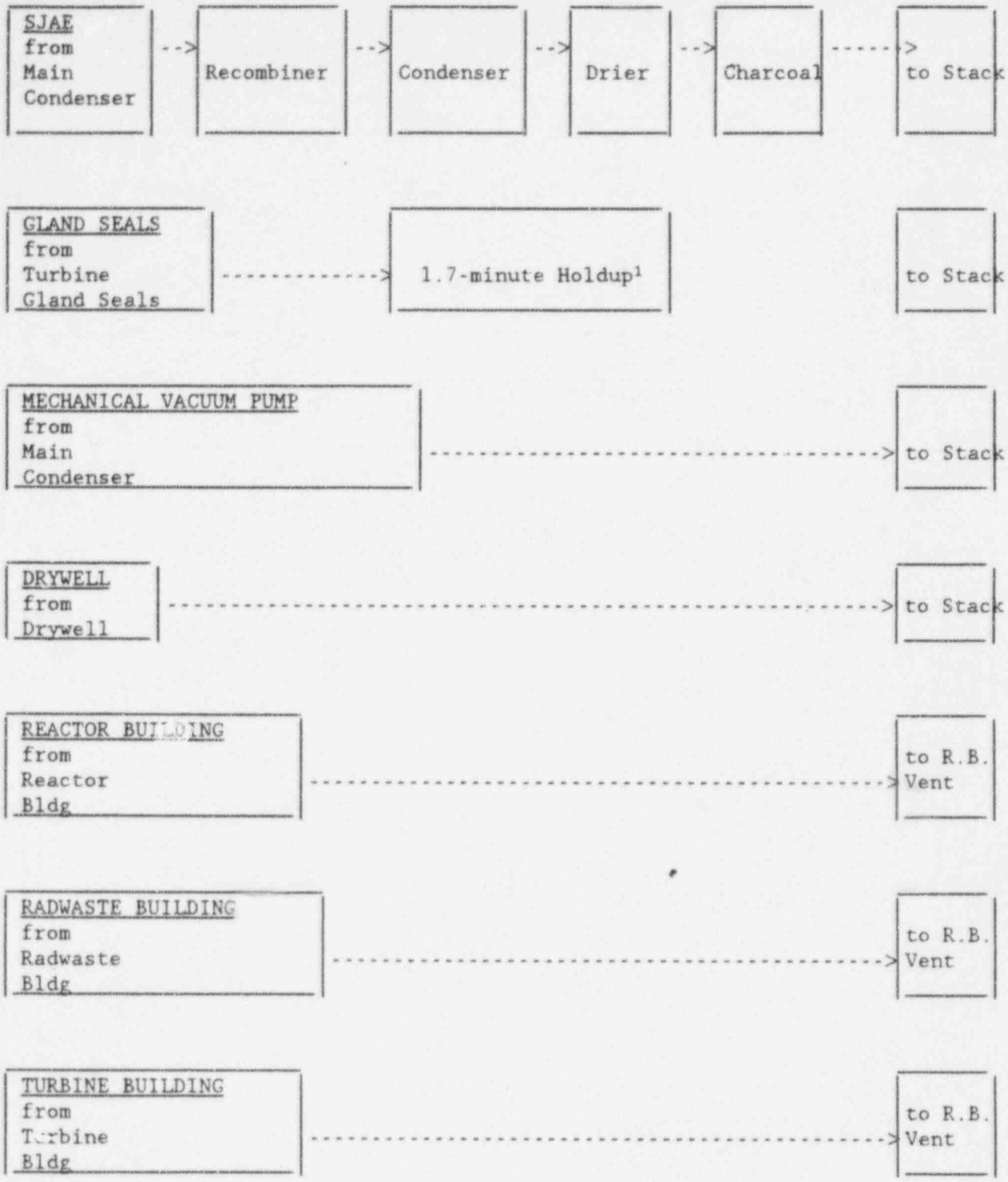


Figure 4.8-1 Liquid Radwaste Treatment System Schematic



[1] No significant effect in reducing offsite doses when compared to transit time required for releases to reach site boundary.

Figure 4.8-2 Gaseous Effluent Treatment Schematic

BASES:

3/4.8 RADIOACTIVE EFFLUENTS

A. Liquid Effluents Concentration

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents at and beyond the site boundary will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in bodies of water at and beyond the site boundary will not result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a member of the public and (2) the limits of 10 CFR Part 20.106(e) to the population.

B. Radioactive Liquid Effluent Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the Offsite Dose Calculation Manual (ODCM) to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

C. Liquid Effluent Treatment

The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criteria 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

D. Gaseous Effluents Dose Rate

This specification is provided to ensure that the dose rate at anytime at and beyond the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20. The annual dose limits are the doses associated with the concentration of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public either within or outside the site boundary to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20.106(b). For members of the public who may at times be within the site boundary, the occupancy of the individual will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding

D. Gaseous Effluents Dose Rate (Cont)

gamma and beta dose rates above background to a member of the public at or beyond the site boundary to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to ≤ 1500 mrem/year for the nearest cow to the plant.

E. Radioactive Gaseous Process and Effluent Monitoring Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The process monitoring instrumentation includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the main condenser offgas treatment system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

F. Gaseous Effluent Treatment

The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

Maintaining the concentration of hydrogen below it's flammability limits provides assurance that releases of radioactive materials will be controlled in conformance with the requirements of General Design Criteria 60 of Appendix A to 10 CFR Part 50.

G. Main Condenser

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to a member of the public at and beyond the site boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

G. Main Condenser (Cont)

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen minute delay before the air ejector off-gas isolation valve is closed. This delay is accounted for by the 30-minute holdup time of the off-gas before it is released to the stack.

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

H. Mechanical Vacuum Pump

The purpose of isolating the mechanical vacuum pump line is to limit the release of activity from the main condenser. During a Control Rod Drop Accident, fission products would be transported from the reactor through the main steam lines to the condenser. The fission product radioactivity would be sensed by the main steam line radioactivity monitors, initiating isolation of the mechanical vacuum pump.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Auxiliary Electrical Equipment

The reactor shall not be made critical unless all of the following conditions are satisfied:

1. At least one offsite transmission line and the startup transformer are available and capable of automatically supplying auxiliary power to the emergency buses.

2. An additional source of offsite power consisting of one of the following:

   a. A transmission line and shutdown transformer capable of supplying power to the emergency 4160 volt buses.

   b. The main transformer and unit auxiliary transformer available and capable of supplying power to the emergency 4160 volt buses.

3. Both diesel generators shall be operable. Each diesel generator shall have a minimum of 19,800 gallons of diesel fuel on site.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Auxiliary Electrical Equipment Surveillance

1. Diesel Generators

   a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one hour period at rated load.

      During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and frequency shall be logged.

   b. Once per operating cycle the condition under which the diesel generator is required will be simulated and test conducted to demonstrate that it will start and accept the emergency load within the specified time sequence. The results shall be logged.

3.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

4.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

A. Auxiliary Electrical Equipment Surveillance (Cont)

1. Verifying de-energization of the emergency buses and load shedding from the emergency buses.

2. Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency buses with permanently connected loads, energizes the auto-connected emergency loads through the load sequence, and operates for ≥ 5 minutes while its generator is loaded with the emergency loads.

   During performance of this surveillance verify that HPCI and RCIC inverters do not trip.

   The results shall be logged.

c. Once per operating cycle with the diesel loaded per 4.9.A.1.b verify that on diesel generator trip, secondary (offsite) AC power is automatically connected within 11.8 to 13.2 seconds to the emergency service buses and emergency loads are energized through the load sequencer in the same manner as described in 4.9.A.1.b.1.

   The results shall be logged.

3.9 AUXILIARY ELECTRICAL SYSTEM

A. Auxiliary Electrical Equipment (Cont)

4. 4160 volt buses A5 and A6 are energized and the associated 480 volt buses are energized.

5. The station and switchyard 125 and 250 volt batteries are operable. Each battery shall have an operable battery charger.

6. Emergency Bus Degraded Voltage Annunciation System as specified in Table 3.2.B.1 is operable.

7. Specification:

Two redundant RPS Electrical Protection Assemblies (EPAs) shall be operable at all times on both inservice power supplies.

Action

a. With one EPA on an inservice power supply inoperable, continued operation is permissible provided that the EPA is returned to operable status or power is transferred to a source with two operable EPAs within 72 hours. If this requirement cannot be met, trip the power source.

b. With both RPS EPAs found to be inoperable on an inservice power supply, continued operation is permissible, provided at least one EPA is restored to operable status or power is transferred to a source with at least one operable EPA within 30 minutes. If this requirement cannot be met, trip the power source.

NOTE: Only applicable if tripping the power source would not result in a scram.

4.9 AUXILIARY ELECTRICAL SYSTEM

A. Auxiliary Electrical Equipment Surveillance (Cont)

d. Once a month the quantity of diesel fuel available shall be logged.

e. Once a month a sample of diesel fuel shall be checked for quality in accordance with ASTM D4057-81 or D4177-82. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-81 and logged.

2. Station and Switchyard Batteries

a. Every week the specific gravity, the voltage and temperature of the pilot cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

c. Once each operating cycle, the stated batteries shall be subjected to a Service Discharge Test (load profile). The specific gravity and voltage of each cell shall be determined after the discharge and logged.

d. Once every five years, the stated batteries shall be subjected to a Performance Discharge Test (capacity). This test will be performed in lieu of the Service Discharge Test requirements of 4.9.A.2.C above.

3.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

B. Operation with Inoperable Equipment

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in a Cold Condition, the availability of electric power shall be as specified in 3.9.B.1, 3.9.B.2, 3.9.B.3, 3.9.B.4, and 3.9.B.5.

1. From and after the date that incoming power is not available from the startup or shutdown transformer, continued reactor operation is permissible under this condition for seven days. During this period, both diesel generators and associated emergency buses must be demonstrated to be operable.

2. From and after the date that incoming power is not available from both startup and shutdown transformers, continued operation is permissible, provided both diesel generators and associated emergency buses are demonstrated to be operable, all core and containment cooling systems are operable, reactor power level is reduced to 25% of design and the NRC is notified within one (1) hour as required by 10CFR50.72.

3. From and after the date that one of the diesel generators or associated emergency bus is made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specification 3.5.F if Specification 3.9.A.1 and 3.9.A.2.a are satisfied.

4. From and after the date that one of the diesel generators or associated emergency buses and either the shutdown or startup transformer power source are

4.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

A. Auxiliary Electrical Equipment Surveillance (Cont)

3. Emergency 4160V Buses A5-A6 Degraded Voltage Annunciation System.

   a. Once each operating cycle, calibrate the alarm sensor.

   b. Once each 31 days perform a channel functional test on the alarm system.

   c. In the event the alarm system is determined inoperable under 3.b above, commence logging safety related bus voltage every 30 minutes until such time as the alarm is restored to operable status.

4. RPS Electrical Protection Assemblies

   a. Each pair of redundant RPS EPAs shall be determined to be operable at least once per 6 months by performance of an instrument functional test.

   b. Once per 18 months each pair of redundant RPS EPAs shall be determined to be operable by performance of an instrument calibration and by verifying tripping of the circuit breakers upon the simulated conditions for automatic actuation of the protective relays within the following limits:

Overvoltage	$\leq$ 132 volts
Undervoltage	$\geq$ 108 volts
Underfrequency	$\geq$ 57H_Z

3.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

B. Operation with Inoperable Equipment (Cont)

made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specification 3.5.F, provided either of the following conditions are satisfied:

a. The startup transformer and both offsite 345 kV transmission lines are available and capable of automatically supplying auxiliary power to the emergency 4160 volt buses.

b. A transmission line and associated shutdown transformer are available and capable of automatically supplying auxiliary power to the emergency 4160 volt buses.

5. From and after the date that one of the 125 or 250 volt battery systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding three days within electrical safety considerations, provided repair work is initiated in the most expeditious manner to return the failed component to an operable state, and Specification 3.5.F is satisfied.

6. With the emergency bus voltage less than 3958.5V but above 3878.7V(excluding transients) during normal operation, transfer the safety related buses to the diesel generators. If grid voltage continues to degrade be in at least Hot Shutdown within the next 4 hours and in Cold Shutdown within the following 12 hours unless the grid conditions improve.

4.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

BASES:

3.9 AUXILIARY ELECTRICAL SYSTEM

The general objective of this Specification is to assure an adequate source of electrical power to operate the auxiliaries during plant operation, to operate facilities to cool and lubricate the plant during shutdown, and to operate the engineered safeguards following an accident. There are three sources of a-c electrical energy available; namely, the startup transformer, the diesel generators and the shutdown transformer. The d-c supply is required for switchgear and engineered safety feature systems. Specification 3.9.A states the required availability of a-c and d-c power; i.e., an active off-site a-c source, a back-up source of off-site a-c power and the maximum amount of on-site a-c and d-c sources.

The diesel fuel supply consists of two (2) 25,000 gallon tanks. Level instrumentation provides operators the information necessary to ensure a minimum supply of 19,800 gallons in each tank.

Auxiliary power for PNPS is supplied from two sources; either the unit auxiliary transformer or the startup transformer. Both of these transformers are sized to carry 100% of the auxiliary load. If the startup transformer is lost, the unit can continue to operate since the unit auxiliary transformer is in service, the shutdown transformer is available, and both diesel generators are operational.

If the startup and shutdown transformers are both lost, the reactor power level must be reduced to a value whereby the unit could safely reject the load and continue to supply auxiliary electric power to the station.

In the normal mode of operation, the startup transformer is energized, two diesel generators and the shutdown transformer are operable. One diesel generator may be allowed out of service based on the availability of power from the startup transformer, the shutdown transformer and the fact that one diesel generator carries sufficient engineered safeguards equipment to cover all breaks. With the shutdown transformer and one diesel generator out of service, both 345kV supply lines must be available for the startup transformer.

Upon the loss of one on-site and one off-site power source, power would be available from the other immediate off-site power source and the one operable on-site diesel to carry sufficient engineered safeguards equipment to cover all breaks. In addition to these two power sources, removal of the Isolated Phase Bus flexible connectors would allow backfeed of power through the main transformer to the unit auxiliary transformer and provide power to carry the full station auxiliary load. The time required to perform this operation is comparable to the time the reactor could remain on RCIC operation before controlled depressurization need be initiated.

BASES:

3.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

A battery charger is supplied with each of the 125 and 250 volt batteries and, in addition, (1) a 125 volt shared back-up battery charger is supplied which can be used for either 125 volt battery, (2) a 250 volt d-c back-up battery charger is supplied. Thus, on loss of normal battery charger, the back-up charger can be used. The 125 volt battery system shall have a minimum of 105 volts at the battery terminals to be considered operable. The 250 volt battery system shall have a minimum of 210 volts at the battery terminals to be considered operable.

Automatic second level undervoltage (Degraded Voltage) protection is installed on the startup transformer and is available when safety related loads are being supplied from this source. During normal operation, the unit auxiliary transformer supplies safety related buses. Automatic second level undervoltage protection is not installed on the unit auxiliary transformer. The Safety Bus Degraded Voltage Alarm System and new Degraded Voltage Operating Procedure will be relied upon to guide Operator action to preclude operation with a degraded bus voltage condition.

Each of the two motor generator sets and the alternate power supply for the Reactor Protection System (RPS) have two Electrical Protection Assemblies (EPAs), installed in series, between the RPS 120 Volt $60H_Z$ power source and its respective RPS bus. A random, or seismically-induced abnormal voltage or frequency condition on the output of an MG Set or on the alternate supply would trip one or both of the EPAs. This protects the RPS components and auxiliaries from damage due to sustained abnormal voltage conditions (overvoltage, undervoltage or underfrequency).

The 72 hour maximum service limit of an "inservice power supply" with only one EPA operable provides reasonable assurance that a single failure would not prevent an abnormal voltage condition from being detected prior to damaging RPS components. The "inservice power supplies" are defined as either the two RPS MG Sets or one RPS MG Set and the alternate power source fed from B10/B6. The 30 minute maximum service limit without either EPA operable provides reasonable assurance that an abnormal voltage condition would not damage RPS components. The tripping of an RPS power supply to an RPS bus will result in a half-scram on that channel.

4.9 AUXILIARY ELECTRICAL SYSTEM

The monthly test of the diesel generator is conducted to check for equipment failures and deterioration. Testing is conducted up to equilibrium operating conditions to demonstrate proper operation at these conditions. The diesel generator will be manually started, connected to the bus, and load picked up. The diesel generator should be loaded to at least 75% of rated load to prevent fouling of the engine. It is expected that the diesel generator will be run for one to two hours. Diesel generator experience at other generating stations indicates that the testing frequency is adequate and provides a high reliability of operation should the system be required.

4.9 AUXILIARY ELECTRICAL SYSTEM (Contd)

Each diesel generator has one air compressor and two air receivers for starting, and one air compressor and three receiver tanks for turbo-charger assist in starting and loading. It is expected that the air compressors will run only infrequently. During the monthly check of the diesel generator, one receiver in each set of receivers will be drawn down below the point at which the corresponding compressor automatically starts to check operation and the ability of the compressors to recharge the receivers.

The diesel generator fuel consumption rate at full load is approximately 193 gallons per hour. Thus, the monthly load test of the diesel generators will test the operation and ability of the fuel oil transfer pumps to refill the day tank and will check the operation of these pumps from the emergency source.

The test of the diesel generator during the refueling outage will be more comprehensive in that it will functionally test the system; i.e., it will check diesel generator starting, closure of the diesel generator breaker, and sequencing of load on the diesel generator. The diesel generator will be started by simulation of a loss of coolant accident. In conjunction with this, an undervoltage condition will be imposed to simulate a loss of offsite power. The timing sequence will be checked to assure that the diesel generators can operate the core spray pumps at rated power within thirty seconds and the LPCI pumps at rated power within forty-three seconds. Additionally, with the diesel generator operating as described above, the capability of supplying power to the emergency bus will be further substantiated. This will be accomplished by tripping the diesel generator breaker and verifying that secondary offsite power is connected to the emergency bus and emergency loads are energized through the load sequence.

The inverters associated with the HPCI and RCIC systems provide power to the flow control mechanisms of these systems. The inverters automatically reset following a high DC input voltage trip. Loss of the RCIC inverter results in a minimum flow condition. Loss of the HPCI inverter results in HPCI going to zero flow. After the inverters reset, RCIC flow will return to normal and HPCI will restart. Demonstrating the inverters do not trip during the once per cycle diesel generator surveillance provides assurance of the successful operation of the circuits, battery chargers, batteries, and inverters.

Periodic tests between refueling outages verify the ability of the diesel generator to run at full load and the core and containment cooling pumps to deliver full flow. Periodic testing of the various components, plus a functional test once per cycle, is sufficient to maintain adequate reliability.

4.9 AUXILIARY ELECTRICAL SYSTEM (Cont)

Although station batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The Service Discharge Test provides indication of the batteries' ability to satisfy the design requirements (battery duty cycle) of the associated dc system. This test will be performed using simulated or actual loads at the rates and for the duration specified in the design load profile. A once per cycle testing interval was chosen to coincide with planned outages.

The Performance Discharge Test provides adequate indication and assurance that the batteries have the specified ampere hour capacity. The results of these tests will be logged and compared with the manufacturer's recommendations of acceptability. This test is performed once every five years in lieu of the Service Discharge test that would normally occur within that time frame.

The diesel fuel oil quality must be checked to ensure proper operation of the diesel generators. Water content should be minimized because water in the fuel could contribute to excessive damage to the diesel engine.

The electrical protection assemblies (EPAs) on the RPS inservice power supplies, either two motor generator sets or one motor generator and the alternative supply, consist of protective relays that trip their incorporated circuit breakers on overvoltage, undervoltage, or underfrequency conditions. There are two EPAs in series per power source. It is necessary to periodically test the relays to ensure the sensor is operating correctly and to ensure the trip unit is operable. Based on experience at conventional and nuclear power plants, a six-month frequency for the channel functional test is established. This frequency is consistent with the Standard Technical Specifications.

The EPAs of the power sources to the RPS shall be determined to be operable by performance of a channel calibration of the relays once per 18 months. During calibration, a transfer to the alternative power source is required; however, prior to switching to alternative feed, de-energization of the applicable MG set power source must be accomplished. This results in a half scram on the channel being calibrated until the alternative power source is connected and the half scram is cleared. Based on operating experience, drift of the EPA protective relays is not significant.

LIMITING CONDITION FOR OPERATION

3.10 CORE ALTERATIONS

Applicability:

Applies to the fuel handling and core reactivity limitations during refueling and core alterations.

Objective:

To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification:

A. Refueling Interlocks

During core alterations when fuel is in the vessel the reactor mode switch shall be locked in the "Refuel" position and the refueling interlocks shall be operable.

B. Core Monitoring

During core alterations when fuel is in the vessel two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability:

Applies to the periodic testing of those interlocks and instrumentation used during refueling and core alterations.

Objective:

To verify the operability of instrumentation and interlocks used in refueling and core alterations.

Specification:

A. Refueling Interlocks

Prior to any fuel handling with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall be tested at weekly intervals thereafter until no longer required. They shall also be tested following any repair work associated with the interlocks.

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response.

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS (Cont)

B. Core Monitoring (Cont)

2. The SRM shall have a minimum of 3 cps except as specified in 3 and 4 below.

3. Prior to spiral unloading, the SRM's shall have an initial count rate of ≥ 3 cps. During spiral unloading, the count rate on the SRM's may drop below 3 cps.

4. During spiral reload, each control cell shall have at least one assembly with a minimum exposure of 1000 MWD/ST.

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 33 feet.

D. Multiple Control Rod Removal

1. Any number of control rods and/or control rod drive mechanisms may be removed from the reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core.

   a. The reactor mode switch is operable and locked in the Refuel position per Specification 3.10.A, except that the Refuel position "one rod out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS (Cont)

B. Core Monitoring (Cont)

Spiral Reload

During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, up to two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these assemblies have loaded, the cps requirement is not necessary.

C. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

D. Multiple Control Rod Removal

1. Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core, verify that:

   a. The reactor mode switch is operable and locked in the Refuel position per Specification 3.10.A.

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS (Cont)

D. Multiple Control Rod Removal (Cont)

b. The source range monitors (SRM) are operable per Specification 3.3.B.4.

c. The Reactivity Margin requirements of Specification 3.3.A.1 are satisfied.

d. All control rods in a 3x3 array centered on each of the control rods being removed are fully inserted and electrically or hydraulically disarmed, or have the surrounding four fuel assemblies removed from the core cell.

e. All other control rods are fully inserted.

f. The four fuel assemblies are removed from the core cell surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS (Cont)

D. Multiple Control Rod Removal (Cont)

b. The SRM channels are operable per Specification 3.3.B.4.

c. The Reactivity Margin requirements of Specification 3.3.A.1 are satisfied.

d. All control rods in 3x3 array centered on each of the control rods removed or being removed are fully inserted and electrically or hydraulically disarmed, or have the surrounding four fuel assemblies removed.

e. All other control rods are fully inserted.

f. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism that is to be removed from the reactor vessel at the same time are removed from the core and/or reactor vessel.

BASES:

3.10 CORE ALTERATIONS

A. Refueling Interlocks

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. The interlocks prevent an inadvertent criticality during refueling operations when the reactivity potential of the core is being altered.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment.

The refueling interlocks include circuitry which senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated which prevent the movement of the refueling equipment or withdrawal of control rods (rod block).

Circuitry is provided which senses the following conditions:

1. All rods inserted.

2. Refueling platform positioned near or over the core.

3. Refueling platform hoists are fuel-loaded (fuel grapple, frame mounted hoist, monorail mounted hoist).

4. Fuel grapple not full up.

5. Service platform hoist fuel-loaded.

6. One rod withdrawn.

When the mode switch is in the "Re-fuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

BASES:

3.10 CORE ALTERATIONS (Cont)

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved ensures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored and ensures startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.

The limiting conditions for operation of the SRM subsystem of the Neutron Monitoring System are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a functional analysis of the neutron monitoring system. The specification is based on the Nuclear Safety Requirements for Plant Operation in Subsection 7.5.10 of the FSAR.

A spiral unloading program is one by which the fuel is in the outermost cells (four fuel bundles surrounding a control blade) is removed first. Unloading continues by removing the remaining outermost fuel cell by cell. The center cell will be the last removed.[1] A spiral loading program is one by which fuel is loaded on the periphery of the previously loaded fueled region beginning around a single SRM. Spiral unloading and reloading will preclude the creation of flux traps (moderator filled cavities surrounded on all sides by fuel).

During spiral unloading, the SRM's shall have an initial count rate of ≥ 3 cps with all rods fully inserted. The count rate will diminish during fuel removal. Under the special condition of complete spiral core unloading, it is expected that the count rate of the SRM's will drop below 3 cps before all of the fuel is unloaded.

Since there will be no reactivity additions, a lower number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, the SRM's will no longer be required. Requiring the SRM's to be operational prior to fuel removal assures that the SRM's are operable and can be relied on even when the count rate may go below 3 cps.

During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, up to two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these assemblies have been loaded, the 3 cps requirement is not necessary.

[1]During selected refueling outages, prior to initiating spiral unloading, the central controlled cell will be removed to facilitate inspection of the Core Spray Spargers.

BASES:

3.10 CORE ALTERATIONS (Cont)

C. Spent Fuel Pool Water Level

To ensure there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (-1 foot) and is well above the level to assure adequate cooling.

D. Multiple Control Rod Removal

These specifications ensure maintenance or repair of control rods or rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with the control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

BASES:

4.10 CORE ALTERATIONS

A. Refueling Interlocks

Complete functional testing of all refueling interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its functions.

B. Core Monitoring

Requiring the SRM's to be functionally tested prior to any core alteration ensures the SRM's will be operable at the start of that alteration. The daily response check of the SRM's ensures their continued operability.

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLY

Applicability:

The Limiting Conditions for Operation associated with fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation with both recirculation pumps operating, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the applicable limiting value specified in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLY

Applicability:

The surveillance requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥ 25% rated thermal power.

3.11 REACTOR FUEL ASSEMBLY (Cont)

B. Linear Heat Generation Rate (LHGR)

During reactor power operation, the LHGR shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

C. Minimum Critical Power Ratio (MCPR)

1. During power operation MCPR shall be ≥ the MCPR operating limit specified in the Core Operating Limits Report. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 REACTOR FUEL ASSEMBLY (Cont)

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.

C. Minimum Critical Power Ratio (MCPR)

1. MCPR shall be determined daily during reactor power operation at > 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

2. The value of τ in Specification 3.11.C.2. shall be equal to 1.0 unless determined from the result of surveillance testing of Specification 4.3.C as follows:

a) τ is defined as

$$\tau = \frac{\tau_{ave} - \tau_B}{1.275 - \tau_B}$$

3.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ration MCPR (Cont'd)

2. The operating limit MCPR values as a function of the τ are given in Table 3.3.1 of the Core Operating Limits Report where τ is given by specification 4.11.C.2.

4.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ration MCPR (Cont'd)

b. The average scram time to the 30% insertion position is determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \quad \tau_i}{\sum_{i=1}^{n} N_i}$$

Where: an n = number of surveillance tests performed to date in the cycle.

N_i = number of active control rods measured in the ith surveillance test.

τ_i = average scram time to the 30% insertion position of all rods measured in the ith surveillance test.

c. The adjusted analysis mean scram time (τ_B) is calculated as follows:

$$\tau_B = \mu + 1.65 \left[\frac{N_i}{\sum_{i=1}^{n} N_i} \right]^{1/2} \sigma$$

Where:

μ = mean of the distribution for average scram insertion time to the 30% position, 0.945 sec.

N_i = total number of active control rod

σ = standard deviation of the distribution for average scram insertion time to the 30% position, 0.064 sec

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLY (Cont)

D. Power/Flow Relationship During Power Operation

The power/flow relationship shall not exceed the limiting values specified in the CORE OPERATING LIMITS REPORT.

If at any time during power operation it is determined by normal surveillance that the limiting value for the power/flow relationship is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the power/flow relationship is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLY (Cont)

D. Power/Flow Relationship During Power Operation

Compliance with the power/flow relationship in Section 3.11.D shall be determined daily during reactor operation.

BASES:

3.11 REACTOR FUEL ASSEMBLY

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50, Appendix K.

The analytical method used to determine the APLHGR limiting values is described in the topical reports listed in Specification 6.9.A.4.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate.

The analytical method used to determine the LHGR limiting value is described in the topical reports listed in Specification 6.9.A.4.

C. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

For any abnormal operating transient analysis with the initial condition of the reactor at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming the instrument trip settings given in Tables 3.1.1, 3.2.A and 3.2.B.

The analytical method used to determine the Operating Limit MCPR values in the CORE OPERATING LIMITS REPORT is described in the topical reports listed in Specification 6.9.A.4. By maintaining MCPR greater than or equal to the Operating Limit MCPR, the Safety Limit MCPR specified in Specification 2.1.2 is maintained in the event of the most limiting abnormal operating transient.

D. Power/Flow Relationship During Power Operation

The power/flow curve is the locus of core thermal power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

BASES:

4.11 REACTOR FUEL ASSEMBLY

A. NA

B. NA

C. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

LIMITING CONDITION FOR OPERATION

3.12 FIRE PROTECTION

Alternate Shutdown Panels

1. Alternate shutdown panels for the following systems shall be OPERABLE:

   1. Core Spray
   2. RHR
   3. RBCCW
   4. Salt Service Water
   5. HPCI
   6. RCIC
   7. Automatic Depressurization
   8. Diesel Generators

APPLICABILITY:

At all times that the system is required to be OPERABLE.

ACTION:

With any of the alternate shutdown panels inoperable,

a. Immediately verify that fire detection with automatic fire suppression for the Cable Spreading Room is Operable. If fire detection with automatic fire suppression cannot be determined operable, within one (1) hour from the time the system is determined to be inoperable, establish a continuous Fire Watch with backup fire suppression.

b. Immediately verify that fire detector zones listed on Table 3.12 are operable for the respective system fire zone(s) for which the panel(s) provided alternate shutdown capability.

   If fire detection zone cannot be determined operable, establish an hourly fire watch patrol to inspect the affected zone(s).

SURVEILLANCE REQUIREMENTS

4.12 FIRE PROTECTION

Alternate Shutdown Panels

The alternate shutdown panels shall be demonstrated to be OPERABLE according to the following:

1. The motor operated valves of the core spray system shall be operated from the alternate shutdown panels once each cycle.

2. The motor operated valves of the RHR system shall be operated once each cycle utilizing the MCC B-17 alternate power source.

3. The pumps of the SSW system shall be operated from the alternate shutdown panels once each cycle.

4. The pumps and motor operated valves of the RBCCW system shall be operated from the alternate shutdown panels once each cycle.

5. Alternate shutdown panel capability for the RCIC and HPCI systems shall be verified to be OPERABLE once each cycle.

6. After each refueling outage and prior to startup, perform a test from the alternate shutdown panel to verify that the relief valve solenoids of the Automatic Depressurization System (ADS) actuate.

7. Once each refueling outage, the diesel generator control circuits shall be isolated from the Cable Spreading Room and the diesel generator started.

TABLE 3.12

FIRE DETECTOR
ZONES ASSOCIATED WITH
ALTERNATE SHUTDOWN PANELS

Alternate Shutdown System	Fire Zone	Detection Panel/Det. Zones
Core Spray	1.1 & .2	C-224/4A
RHR	1.1 & .2	C223/3C
RBCCW	1.21 & .22	C-222/2A & 2B
SSW	5.1 & .2 & .3	N/A
HPCI	1.3 & .4	C-223/3D & 3E
RCIC	1.5	C-223/3A & 3B
ADS	1.1 & .2	C-224/4A
DGS	4.1 & .3	C-93/1 & 2

BASES:

3/4.12 FIRE PROTECTION

The alternate shutdown system, independent of cabling and equipment in the Cable Spreading Room is provided to effect safe shutdown of Pilgrim in the event of a fire in the Cable Spreading Room. This is accomplished by installing isolation switches for safety-related equipment that will provide the capability for the plant operators to reach a safe shutdown condition. These switches will isolate their associated equipment from the CSR cables, thus transfer control from the Control Room to the local emergency shutdown stations outside the CSR. These isolation switches are located in alternate shutdown panels and are located as close as practical to the equipment or switchgear they serve.

An emergency shutdown procedure, which is compatible with the design modifications and plant operator availability, provides step-by-step actions to initiate safe shutdown operation. Operator actions to isolate safety-related cables passing through the CSR is initiated as soon as a fire which is not immediately extinguishable is detected and confirmed in the CSR.

Alternate shutdown panels are provided for the following systems:

a. Core Spray

b. RHR

c. RBCCW

d. Salt Service Water

e. HPCI

f. RCIC

g. Automatic Depressurization System

h. Diesel Generators

Inoperability of the above listed systems does not require entry into LCO action statements for the alternate shutdown panels.

A surveillance frequency of once per cycle is considered prudent and more frequent testing not warranted. The frequency of once per refueling outage for testing the diesel generators prevents unnecessarily rendering them inoperable during normal power operation. The frequency of once per refueling outage for the Automatic Depressurization System is consistent with the existing surveillance frequency for this system. Requiring this surveillance to be performed during a refueling outage will also assure that plant conditions will allow for safe access to the ADS solenoids.

LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE CODE TESTING

Applicability:

Applies to ASME Code Class 1, 2 and 3 or equivalent pumps and valves.

Objective:

To assure the operational readiness of ASME Code Class 1, 2, and 3 (Safety Related) or equivalent (important to safety) pumps and valves.

Specification:

A. Inservice Code Testing of Pumps and Valves

1. Based on the Facility Commercial Operation Date, Inservice Code Testing of safety and safety-related pumps and valves shall be performed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI "Rules for Inservice Testing of Nuclear Power Plant Components" Subsections IWP and IWV as required by 10CFR50.55a(f), except where specific relief has been granted by the NRC pursuant to 10CFR50.55a(f)(6)(i).

SURVEILLANCE REQUIREMENTS

4.13 INSERVICE CODE TESTING

Applicability:

Applies to the periodic testing requirements of ASME Code Class 1, 2 and 3 or equivalent pumps and valves.

Objective:

To assess the operational readiness of safety and safety-related pumps and valves by performance of inservice tests.

Specification:

A. Inservice Code Testing of Pump and Valves

1. Inservice Code Testing activities shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50.55a(f), with the exemptions and alternate testing that have been approved by the NRC pursuant to 10CFR50.55a(f)(6)(i). These exemptions and alternate testing are included in the PNPS Inservice Testing Program.

2. Test Frequencies for Code Terminology when performing Inservice Test activities.

Code Terminology	Frequencies
Weekly	7 Days
Monthly	31 Days
Quarterly or 3 Mths	92 Days
Semiannually/ 6 Mths	184 Days
9 Months	276 Days
Yearly/Annually	366 Days
Biannual/2 Yrs	732 Days

LIMITING CONDITIONS FOR OPERATION

3.13 INSERVICE CODE TESTING

SURVEILLANCE REQUIREMENTS

4.13 INSERVICE CODE TESTING

3. The provi. ns in Definitions (1.0) for REFUEL 3 INTERVAL, SURVEILLANCE FREQUENCY, and SURVEILLANCE INTERVAL are applicable to Code testing and to the above frequencies for performing Code testing activities.

4. Performance of Code testing shall be in addition to other specified Surveillance Requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall supersede the requirements of Technical Specifications.

BASES:

3.13 and 4.13 Inservice Code Testing

The Limiting Conditions for Operation establishes the requirement that inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the periodically updated edition of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10CFR50, Section 50.55a(g). These requirements apply except when relief has been requested pursuant to 10CFR50.55a(g)(6)(i) and granted by the NRC. The NRC may grant relief pursuant to 10CFR50.55a(a)(3)(i), 10CFR50.55a(a)(3)(ii) or 10CFR50.55a(a)(6)(i).

The detailed procedures for testing of pumps and valves are documented in the PNPS Inservice Testing Program.

This specification includes a clarification of the frequencies for performing the testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in Surveillance Frequencies throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice testing activities.

Under the terms of this Specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example:

- Technical Specifications require components to be declared operable prior to entry into an operational mode. The ASME B&PV Code provision which allows pumps and valves to be tested up to one week after return to normal operation is superseded (and not allowed) by the more restrictive requirements of Technical Specifications.

- The allowance for a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable is superseded (and not allowed) by the more restrictive Technical Specification definition of operability which does not allow a grace period.

4.0 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

4.1 SEALED SOURCE CONTAMINATION

Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination at all times.

Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

A. Either decontaminated and repaired, or

B. Disposed of in accordance with Commission Regulations.

4.2 SURVEILLANCE REQUIREMENTS

A. Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

1. The licensee, or

2. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

B. Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

1. Sources in use - At least once per six months for all sealed sources containing radioactive material:

a. With a half-life greater than 30 days, excluding Hydrogen 3, and

b. In any form other than gas.

2. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

4.0 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

4.2 SURVEILLANCE REQUIREMENTS (Cont)

B. Test Frequencies (Cont)

3. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.3 REPORTS

A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of ≥ 0.005 microcuries of removable contamination.

4.4 RECORDS RETENTION

A complete inventory of radioactive materials in possession shall be maintained current at all times.

Records required to be maintained for two years:

A. Test results, in units of microcuries, for leak tests performed pursuant to Specification 4.2.

B. Record of annual physical inventory verifying accountability of sources on record.

BASES:

SEALED SOURCE CONTAMINATION

The limitations on sealed sourced removable contamination ensure that the body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitation on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Quantities of interest to this specification which are exempt from the leakage testing are consistent with the criteria of 10 CFR Part 30.11-20 and 70.19. Leakage from sources excluded from the requirements of this specification is not likely to represent more than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Pilgrim Nuclear Power Station is located on the Western Shore of Cape Cod Bay in the Town of Plymouth, Plymouth County, Massachusetts. The site is located at approximately 41°51' north latitude and 70°35' west longitude on the Manomet Quadrangle, Massachusetts, Plymouth County 7.5 Minute Series (topographic) map issued by U.S. Geological Survey. UTM coordinates are 19-46446N-3692E.

The reactor (center line) is located approximately 1800 feet from the nearest property boundary.

5.2 REACTOR CORE

The reactor vessel core design shall be as described in the CORE OPERATING LIMITS REPORT and shall be limited to those fuel assemblies which have been analyzed with NRC-approved codes and methods and approved by the NRC in its acceptance of Amendment 22 of GESTAR II.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2.2 of the FSAR. The applicable design codes shall be as described in Table 4.2.1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2.1 of the FSAR. The applicable design codes shall be as described in Section 12.2.2.8 of the FSAR.

B. The secondary containment shall be as described in Section 5.3.2 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.3.4 of the FSAR.

5.5 FUEL STORAGE

A. The new fuel storage facility shall be such that the dry K_{eff} is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

C. Each fuel assembly in the spent fuel pool shall have a maximum K-infinity less than or equal to 1.32 and an enrichment of 4.6% U-235 or less averaged over the axial planar zone of highest average enrichment.

D. The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 3859.

E. Loads in excess of 2000 lbs. shall be prohibited from travel over fuel assemblies in the spent fuel storage pool.

F. No fuel which has decayed for less than 200 days shall be stored in racks within an arc described by the height of the cask around the periphery of the energy absorbing pad.

5.6 SEISMIC DESIGN

The station Class I structures and systems have been designed for ground accelerations of 0.08g (design earthquake) and 0.15g (maximum credible earthquake).

BASES:

5.5 FUEL STORAGE

The fuel storage assembly K-infinity in Section 5.5.C refers to the maximum K-infinity for the standard reactor core geometry. Storage of fuel assemblies meeting specification 5.5.C will result in K_{eff} less than 0.95, for both normal and abnormal storage conditions.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

The Station Director shall be accountable for overall facility operation. In his absence, the Station Director shall designate, in writing, the individual to assume this responsibility.

6.2 ORGANIZATION

A. Offsite and Onsite Organizations

Onsite and Offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Pilgrim Station Final Safety Analysis Report.

2. The Station Director shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

3. The Senior Vice President - Nuclear shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

B. Unit Staff

1. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

6.2 ORGANIZATION (Cont)

B. Unit Staff (Cont)

2. When the unit is in an operational mode other than cold shutdown or refueling, a person holding a Senior Reactor Operator License shall be present in the control room at all times. In addition to this Senior Operator, a Licensed Operator or Senior Operator shall be present at the controls when fuel is in the vessel.

3. At least two Licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

4. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

5. ALL CORE ALTERATIONS performed while fuel is in the reactor vessel after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

6. Deleted

7. The Chief Operating Engineer, Nuclear Watch Engineers, and Nuclear Operations Supervisors shall hold a Senior Reactor Operator License. The Nuclear Plant Operators shall hold a Reactor Operator License.

6.3 UNIT STAFF QUALIFICATIONS

The qualifications with regard to educational and experience backgrounds of the unit staff at the time of appointment to the active position shall meet the requirements as described in the American National Standards Institute N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants." In addition, the individual performing the function of Radiation Protection Manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September, 1975.

6.4 TRAINING

A retraining and replacement training program for the unit staff shall be maintained under the direction of the Nuclear Training Department Manager. The training programs for the licensed personnel shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10CFR Part 55.

6.5 REVIEW AND AUDIT

A. Operations Review Committee (ORC)

1. Function

   The ORC shall function to advise the Station Director on all matters related to safety.

2. Composition

   The ORC shall be composed of a Chairman, and at least six members, who shall be appointed in writing by the Station Director from senior experienced onsite individuals, at the manager level or equivalent, representing each of the following disciplines: plant operations, plant maintenance, plant technical, reactor engineering, radiation protection, and chemistry.

3. Alternates

   Alternates shall be appointed in writing by the Station Director to serve on a temporary basis.

4. Meeting Frequency

   The ORC shall meet at least once per calendar month and as convened by the ORC Chairman.

5. Quorum

   A quorum of the ORC shall consist of the Chairman or designated alternate and a majority of members/designated alternates; however, no more than two alternates shall be allowed to meet quorum requirements.

6. Responsibilities

   The ORC shall be responsible for:

   a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto that affect nuclear safety.

   b. Review all proposed tests and experiments that affect nuclear safety.

   c. Review of all proposed changes to the Technical Specifications.

   d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

6.5 REVIEW AND AUDIT (Cont)

A. Operations Review Committee (ORC) (Cont)

6 Responsibilities (Cont)

e. Review of facility operations to detect potential safety hazards.

f. Review of the Station Security Plan and implementing procedures and changes to the plan and procedures.

g. Review of the Emergency Plan and implementing procedures and changes to the plan and procedures.

h. Performance of special reviews and investigations and reports thereon as requested by the Nuclear Safety Review and Audit Committee (NSRAC) Chairman.

i. Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Station Director, the NSRAC Chairman, and the Senior Vice President - Nuclear.

j. Review the Station Fire Protection Program and implementing procedures and changes to the Program and implementing procedures.

The ORC Chairman may appoint subcommittees composed of personnel who are not members of ORC to perform staff work necessary to the efficient functioning of ORC.

7. Authority

a. Recommend in writing to the Station Director the approval or disapproval of items considered under 6.5.A.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.A.6(a) through (d) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Station Director, the Nuclear Safety Review and Audit Committee, and the Senior Vice President - Nuclear of disagreement between the ORC Members and the ORC Chairman. The Station Director shall have responsibility for resolution of such disagreements.

8. Records

The ORC shall maintain written minutes of each meeting and copies shall be forwarded to the Station Director and the NSRAC Chairman.

6.5 REVIEW AND AUDIT (Cont)

B. Nuclear Safety Review and Audit Committee (NSRAC)

1. Function

The NSRAC shall function to provide independent review and audit of designated activities in the areas of:

1. nuclear power plant operations;
2. nuclear engineering;
3. chemistry and radiochemistry;
4. metallurgy;
5. instrumentation and control;
6. radiological safety;
7. mechanical and electrical engineering;
8. quality assurance practices;
9. fire protection.

2. Composition

The NSRAC Chairman and other members shall be appointed by the Senior Vice President - Nuclear.

The membership shall collectively possess a broad based level of experience and competence enabling the Committee to review and audit those activities designated in 6.5.B.1 above and to recognize when it is necessary to obtain technical advice and counsel. The collective competence of the Committee will be maintained as changes to the membership are made. The membership shall consist of a minimum of five persons of whom no more than a minority are members of the plant staff.

3. Alternatives

Alternate members shall be appointed in writing by the Senior Vice President - Nuclear or the Chairman to serve on a temporary basis.

4. Consultants

Consultants shall be utilized as determined by the NSRAC Chairman to provide expert advice to the NSRAC.

5. Meeting Frequency

The NSRAC shall meet at least once per six months.

6. Quorum

A quorum of the NSRAC shall consist of the Chairman or his designated alternate and at least four NSRAC members including alternates. The Vice Chairman is the designated alternate to the Chairman. No more than a minority of the quorum shall have line responsibility for operation of the facility. No more than two alternates shall participate in a quorum at any one time.

6.5 REVIEW AND AUDIT (Cont)

B. Nuclear Safety Review and Audit Committee (NSRAC) (Cont)

7. Review

The NSRAC shall review:

a. The written safety evaluations for:

1. changes in the facility as described in the Final Safety Analysis Report;

2. changes in the procedures described in Chapter 13 of the Final Safety Analysis Report;

3. test and experiments not described in the Final Safety Analysis Report

to verify that such changes, tests or experiments did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59(a)(2).

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or operating license.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that effect nuclear safety.

g. All events which are required by 10 CFR 50.73 to be reported to the NRC in writing.

h. Any other matter involving safe operation of the nuclear plant which NSRAC deems appropriate for consideration or which is referred to NSRAC for the onsite operating organization or by other functional organizational units within Boston Edison.

i. Reports and meeting minutes of the Operations Review Committee.

6.5 REVIEW AND AUDIT (Cont)

B. Nuclear Safety Review and Audit Committee (NSRAC) (Cont)

8. Audits

Audits of facility activities shall be performed under the cognizance of the NSRAC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The training and qualifications of the entire unit staff at least once per year.

c. The results of all actions required by deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per two years.

e. The Emergency Plan and implementing procedures at least once per two years.

f. The Station Security Plan and implementing procedures at least once per two years.

g. Any other area of facility operation considered appropriate by the NSRAC or the Senior Vice President - Nuclear.

h. The Fire Protection Program and implementing procedures at least once per two years.

9. Authority

The NSRAC shall report to and advise the Senior Vice President - Nuclear on those areas of responsibility specified in Section 6.5.B.7 and 6.5.B.8.

10. Records

Records of NSRAC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSRAC meeting shall be prepared, approved and forwarded to the Senior Vice President - Nuclear, NSRAC members, and others the Chairman may designate.

6.5 REVIEW AND AUDIT (Cont)

B. Nuclear Safety Review and Audit Committee (NSRAC) (Cont)

10. Records (Cont)

b. Reports of reviews encompassed by Section 6.5.B.7.e, f, g and h above, shall be prepared, approved and forwarded to the Senior Vice President - Nuclear, with a copy to the Station Director within 21 days following the completion of the review.

c. Audit reports encompassed by Section 6.5.B.8 above shall be forwarded to the Senior Vice President - Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENT ACTION

The following actions shall be taken for each reportable event:

A. The Commission shall be notified and/or a report submitted pursuant to the requirements of either 10 CFR 50.72 or 10 CFR 50.73.

B. Each Reportable Event Report submitted to the Commission shall be reviewed by the ORC and submitted to the NSRAC Chairman and the Station Director.

6.7 Deleted

6.8 PROCEDURES

A. Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7 - 1972 and Appendix "A" of USNRC Regulatory Guide 1.33, except as provided in 6.8.B and 6.8.C below.

B. Each procedure of 6.8.A above, and changes thereto, shall be reviewed by the ORC and approved by the responsible department manager prior to implementation. These procedures shall be reviewed periodically as set forth in administrative procedures.

NOTE: ORC review and approval of procedures for vendors/contractors, who have a QA Program approved by Boston Edison Company, is not required for work performed at the vendor/contractor facility.

C. Temporary changes to procedures of 6.8.A above may be made provided:

1. The intent of the original procedure is not altered.

6.8 PROCEDURES (Cont)

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's license on the unit affected.

3. The change is documented, subsequently reviewed by the ORC within 14 days of implementation, and approved by the responsible department manager.

D. Written procedures to implement the Fire Protection Program shall be established, implemented and maintained.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Commission.

A. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

6.9 REPORTING REQUIREMENTS

A. Routine Reports (Cont)

2. Monthly Operating Report

Routine reports of operating statistics, shutdown experience and forced reductions in power shall be submitted on a monthly basis to the Commission to arrive no later than the 15th of each month following the calendar month covered by the report.

The Monthly Operating Report shall include a narrative summary of operating experience that describes the operation of the facility, including safety-related maintenance, for the monthly report period.

3. Occupational Exposure Tabulation

A tabulation of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g. reactor operations and surveillance inservice inspection, routine maintenance, special maintenance (including a description), waste processing, and refueling shall be submitted on an annual basis. This tabulation supplements the requirements of 20.407 of 10 CFR 20. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

4. Core Operating Limits Report

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (the approved version at the time the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT), in NEDC-31852P, "Pilgrim Nuclear Power Station SAFER/GESTR-LOCA Loss of Coolant Accident Analysis", dated September, 1990 (the approved version at the time the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT), and in NEDC-31312-P, "ARTS Improvement Program Analyses for Pilgrim Nuclear Power Station", dated September 1987, (the approved version at the time the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT).

6.9 REPORTING REQUIREMENTS

A. Routine Reports (Cont)

4. Core Operating Limits Report (Cont)

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Deleted

C. Unique Reporting Requirements

1. Semi-Annual Radioactive Effluent Release Report

a. Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The report should be in accordance with Appendix B of Regulatory Guide 1.21 (Revision 145 1) dated June, 1974. A supplemental report containing dose assessments for the previous year shall be submitted annually within 90 days after January 1.

b. Any changes to the Offsite Dose Calculation Manual (ODCM) shall be submitted to the Commission in the Semi-Annual Radioactive Effluent Release Report.

2. Annual Radiological Environmental Monitoring Report

A report on the radiological environmental surveillance program for the previous calendar year of operation shall be submitted to the Commission prior to May 1 of the year. The reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, operational controls and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of any land use surveys which affect the choice of sample locations. If harmful effects or evidence of irreversible damage are detected by the monitoring, the licensee shall provide an analysis of the problem and a proposed course of action to alleviate the problem.

6.9 REPORTING REQUIREMENTS (Cont)

C. Unique Reporting Requirements (Cont)

2. Annual Radiological Environmental Monitoring Report (Cont)

The Annual Radiological Environmental Monitoring Report shall include a summary of the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the Offsite Dose Calculation Manual (ODCM) as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 145 1, November 1979.

In the event that some results are not available prior to May 1 of the year, the report shall be submitted, noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The report shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps[1] covering all sampling locations keyed to a table giving distances and directions from the centerline of the reactor; discussion of all deviations from the sampling schedule of Table 8.1-1; and discussion of all analyses in which the lower limits of detection (LLD) required by Table 8.1-4 were not achievable.

3. Special Reports

Special reports shall be submitted as indicated in Table 6.9.1.

6.10 RECORD RETENTION

A. The following records shall be retained for at least five years:

1. Records of facility operation covering time interval at each power level.

2. Records of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

3. Reportable Event Reports.

4. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

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[1] One map shall cover stations near the site boundary; a second shall include the more distant stations.

6.10 RECORD RETENTION (Cont)

5. Records of reactor tests and experiments.

6. Records of changes made to Operating Procedures.

7. Records of radioactive shipments.

8. Records of sealed source leak tests and results.

9. Records of annual physical inventory of all source material of record.

B. The following records shall be retained for the duration of the Operating License:

1. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

2. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

3. Records of facility radiation and contamination surveys.

4. Records of radiation exposure for all individuals entering radiation control areas.

5. Records of the service lives of all hydraulic and mechanical snubbers listed in PNPS procedures including the date at which the service life commences and associated installation and maintenance records.

6. Records of gaseous and liquid radioactive material released to the environs.

7. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

8. Records of training and qualification for current members of the plant staff.

9. Records of in-service inspections performed pursuant to these Technical Specifications.

10. Records of Quality Assurance activities required by the QA Manual.

11. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

12. Records of meetings of the ORC and the NSRAC.

13. Records for Environmental Qualification.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 Deleted

6.13 HIGH RADIATION AREA

1. In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Manager in the Radiation Work Permit.

2. The requirements of 6.13.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Nuclear Watch Engineer on duty and/or the Radiation Protection Manager.

* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirements during the performance of this assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

6.14 FIRE PROTECTION PROGRAM

The following inspections and audits shall be performed:

1. An independent fire protection inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

2. An inspection and audit by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

TABLE 6.2-1

MINIMUM OPERATING SHIFT CREW COMPOSITION

TECHNICAL SPECIFICATION

STATION CONDITION	CREW (a)	MINIMUM NUMBER ON DUTY
OPERATING	Licensed Senior Reactor Operator	2 (b)
	Licensed Reactor Operator	2
	Unlicensed Operator	2
	Shift Technical Advisor	1
COLD SHUTDOWN and REFUELING	Licensed Senior Reactor Operator	1
	Licensed Reactor Operator	1
	Unlicensed Operator	1
	Shift Technical Advisor	None Required

Notes:

(a) Higher grade licensed operators may take the place of lower grade licensed or unlicensed personnel.

(b) A Shift Technical Advisor (STA) with a Senior Reactor Operator license may simultaneously serve as STA and SRO.

TABLE 6.9-1

REPORTS

	Area	Reference	Submittal Date
a.	Secondary Containment Leak Rate Testing (1)	4.7.C.1.c	Upon completion of each test (2)
b.	(Deleted)		
c.	(Deleted)		
d.	(Deleted)		
e.	Standby Liquid Control solution enrichment out of specification	3.4.C.3	Fourteen days after receipt of a non-complying enrichment report or lack of receipt of such a report within the required thirty days, if enrichment compliance cannot be achieved within seven days.

NOTES:

1. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report shall include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

2. The report shall be submitted approximately 90 days after completion of each test. Test periods shall be based on the commercial service date as the starting point.

OPERATIONAL OBJECTIVES

7.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

7.1 MONITORING PROGRAM

Applicability:

At all times.

Specification:

A. Environmental Monitoring

An environmental monitoring program shall be conducted to evaluate the effects of station operation on the environs and to verify the effectiveness of the source controls on radioactive materials.

The radiological environmental monitoring program shall be conducted as specified in Table 8.1-1.

Action:

1. With the radiological environmental monitoring program not being conducted as specified in Table 8.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Monitoring Report required by Specification 6.9.C.2, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

2. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 7.1-1 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, a

SURVEILLANCE REQUIREMENTS

8.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

8.1 MONITORING PROGRAM

Specification:

A. Environmental Monitoring

The radiological environmental monitoring samples shall be collected pursuant to Table 8.1-1 from the specific locations given in the table and figure(s) in the Offsite Dose Calculation Manual (ODCM) and shall be analyzed pursuant to the requirements of Table 8.1-1 and the detection capabilities required by Table 8.1-4.

1. Cumulative dose contributions for the current calendar year from radionuclides detected in environmental samples shall be determined in accordance with the methodology and parameters in the ODCM. These results will be reported in the Annual Radiological Environmental Monitoring Report.

7.1 MONITORING PROGRAM (Cont)

A. Environmental Monitoring (Cont)

special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 7.2, 7.3, and 7.4. When more than one of the radionuclides in Table 7.1-1 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration(1)}}{\text{reporting level(1)}} + \frac{\text{concentration(2)}}{\text{reporting level(1)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 7.1-1 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a member of the public is equal to or greater than the calendar year limits of Specifications 7.2, 7.3, and 7.4. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Monitoring Report.

3. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 8.1-1, identify locations for obtaining replacement samples and add them to the Radiological Environmental Monitoring Program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program.

8.1 MONITORING PROGRAM (Cont)

7.1 MONITORING PROGRAM (Cont)

A. Environmental Monitoring (Cont)

Pursuant to Specification 6.9.C.2, identify the cause of the unavailability of samples and identify the new location(s) obtaining replacement samples in the next Annual Environmental Radiation Monitoring Report and also include in the report the table for the ODCM reflecting the new location(s).

B. Land Use Census

A land use census shall be conducted and shall identify, within a distance of 8 km (5 miles), the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify, within a distance of 5 km (3 miles), the locations in each of the 16 meteorological sectors of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation.

Action

1. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 8.4.A, identify the new location(s) in the next Annual Environmental Radiological Monitoring Report.

8.1 MONITORING PROGRAM (Cont)

B. Land Use Census

The land use census shall be conducted during the growing season, at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Monitoring Report.

Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of the two different direction sectors with the highest predicted D/Qs, in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 8.1-1 shall be followed, including analysis of control samples.

7.1 MONITORING PROGRAM (Cont)

B Land Use Census (Cont)

2. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 7.1, add the new location(s) to the Radiological Environmental Monitoring Program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Identify the new location(s) in the next Annual Environmental Radiological Monitoring Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

7.2 DOSE - LIQUIDS

Applicability:

At all times.

Specification:

A. The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released at and beyond the site boundary shall be limited:

1. During any calendar quarter, to $\leq$ 1.5 mrem to the total body and to $\leq$ 5 mrem to any organ, and

8.1 MONITORING PROGRAM (Cont)

B Land Use Census (Cont)

8.2 DOSE - LIQUIDS

Specification:

A. Dose Calculations - Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM for each calendar month during which releases occurred.

7.2 DOSE - LIQUIDS (Cont)

2. During any calendar year to ≤ 3 mrem to the total body and to ≤ 10 mrem to any organ.

Action

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, a special report that identifies the cause(s), corrective actions taken, and corrective actions to be taken.

7.3 DOSE - NOBLE GASES

Applicability:

At all times.

Specification:

A. The air dose in areas at and beyond the site boundary due to noble gases released in gaseous effluents shall be limited to the following:

1. During any calendar quarter, to ≤ 5 mrad for gamma radiation and ≤ 10 mrad for beta radiation; and

2. During any calendar year, to ≤ 10 mrad for gamma radiation and ≤ 20 mrad for beta radiation.

Action

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, a special report which identifies the cause(s), the corrective actions taken, and corrective actions to be taken.

8.2 DOSE - LIQUIDS (Cont)

8.3 DOSE - NOBLE GASES

Specification:

A. Dose Calculations - Cumulative dose contributions for the total time period shall be determined in accordance with the ODCM for each calendar month during which releases occurred.

7.4 DOSE - IODINE-131, IODINE-133, RADIOACTIVE MATERIAL IN PARTICULATE FORM, AND TRITIUM

Applicability:

At all times

Specification:

A. The dose to a member of the public from iodine-131, iodine-133, radioactive materials in particulate form with half-lives greater than 8 days, and tritium in gaseous effluents released to areas at and beyond the site boundary shall be limited to the following:

1. During any calendar quarter to ≤ 7.5 mrem to any organ, and

2. During any calendar year to ≤ 15 mrem to any organ.

Action

With the calculated dose from the release of iodine-131, iodine-133, radioactive materials in particulate form, and tritium in gaseous effluents exceeding any of the above limits; prepare and submit to the Commission within 30 days, a special report which identifies the cause(s), corrective actions taken, and the corrective actions to be taken.

7.5 TOTAL DOSE

Applicability:

At all times.

Specification:

A. The dose or dose commitment to any member of the public from Pilgrim Station sources is limited to ≤ 25 mrem to the total body or any organ (except the thyroid, which

8.4 DOSE - IODINE-131, IODINE-133, RADIOACTIVE MATERIAL IN PARTICULATE FORM, AND TRITIUM

Specification:

A. Dose Calculations - Cumulative dose contributions for the total time period shall be determined for iodine-131, iodine-133, radioactive material in particulate form with half-lives greater than 8 days, and tritium in accordance with the ODCM for each calendar month during which releases occurred.

8.5 TOTAL DOSE

Specification:

A. Dose Calculations - Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 7.2.A, 7.3.A, and 7.4.A; and in accordance with the ODCM.

7.5 TOTAL DOSE (Cont)

is limited to ≤ 75 mrem) over a period of any calendar year.

Action

With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 7.2.A, 7.3.A, or 7.4.A; prepare and submit a special report to the Commission and limit the subsequent releases such that the dose or dose commitment to any member of the public from all uranium fuel cycle sources is limited to ≤ 25 mrem to the total body or any organ (except thyroid, which is limited to ≤ 75 mrem) over any calendar year. This special report shall include an analysis which demonstrates that radiation exposures to all members of the public from all uranium fuel cycle sources (including all effluent pathways and direct radiation) are less than the 40 CFR, Part 190 standard. Otherwise, obtain a variance from the Commission to permit releases which exceed the 40 CFR, Part 190 standard.

8.5 TOTAL DOSE (Cont)

TABLE 7.1-1

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

	Reporting Levels				
Analysis	Water (pCi/L)	Airborne Particulate or Gases (pCi/M^3)	Wet Solids (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	2×10^4				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-95	4×10^2				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-140	2×10^2			3×10^2	

PNPS
TABLE 8.1-1

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway or Sample Type	Locations (Direction-Distance) from Reactor	Sampling and Collection Frequency	Type and Frequency of Analysis
AIRBORNE			
Particulates	11 Locations (See Table 8.1-2)	Continuous sampling over one week	Gross beta radioactivity 24 hours or more after filter change[1]
Quarterly	11 Locations (See Table 8.1-2)		Composite (by location) for gamma isotopic[2]
Radioiodine	11 Locations (See Table 8.1-2)	Continuous sampling with canister collection weekly	Analyze weekly for I-131
DIRECT[9]	40 Locations (See Table 8.1-3)	Quarterly	Gamma exposure quarterly
	Plymouth Beach and Priscilla/White Horse Beach	Annually	Gamma exposure survey[9]
WATERBORNE (Surface Water)	Discharge Canal	Continuous composite sample	Gamma isotopic[2] monthly, and composite for H-3 analysis quarterly[3]
	Bartlett Pond (SE-1.7 mi)	Weekly grab sample	
	Powder Point (NNW-7.8 mi)[4]	Weekly grab sample	

TABLE 8.1-1 (Cont)

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway or Sample Type	Locations (Direction-Distance) from Reactor	Sampling and Collection Frequency	Type and Frequency of Analysis
AQUATIC			
Shellfish (clams, mussels or quahogs as available)	Discharge Outfall Duxbury Bay Manomet Point Plymouth of Kingston Harbor Marshfield[4]	Quarterly (at approximate 3-month intervals)	Gamma isotopic[2], [6]
Lobster	Vicinity of discharge point Offshore[4]	Four times per season Once per season	Gamma isotopic[2] on edible portions
Fish	Vicinity of discharge point Offshore[4]	Quarterly (when particular species available) for Groups I and II[5], in season for Groups III and IV[5] annually for each group	Gamma isotopic[2] on edible portions[5]
Sediments	Rocky Point Plymouth Harbor Duxbury Bay Plymouth Beach Manomet Point Marshfield	Semiannually	Gamma isotopic [2,3,7]

TABLE 8.1-1 (Cont)

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway or Sample Type	Locations (Direction-Distance) from Reactor	Sampling and Collection Frequency	Type and Frequency of Analysis
INGESTION (Terrestrial)			
Milk	Plymouth County Farm when available (W-3.5 mi)[8] Whitman Farm (NW-21 mi)[4]	Semimonthly during periods when animals are on pasture, otherwise monthly	Gamma isotopic[2], radio-iodine analysis all samples
Cranberries	Manomet Point Bog (SE-2.6 mi) Bartlett Rd. Bog (SSE/S-2.8mi) Pine St. Bog (WNW-17 mi)[4]	At time of harvest	Gamma isotopic[2], on edible portions
Tuberous and green leafy vegetables	Plymouth County Farm (W-3.5 mi)[8] Bridgewater Farm (W-20 mi)[4]	At time of harvest	Gamma isotopic[2] on edible portions
Beef Forage	Plymouth County Farm (W-3.5 mi)[8] Whitman Farm (NW-21 mi)[4]	Annually	Gamma isotopic[2]

NOTES FOR TABLE 8.1-1

1 If gross beta radioactivity is greater than 10 times the control value, gamma isotopic will be performed on the sample.

2 Gamma isotopic means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

3 If integrated gamma activity (less K-40) is greater than 10 times the control value (less K-40), strontium-90 analysis will be performed on the sample.

4 Indicates control location.

5 Fish analyses will be performed on a minimum of 2 sub-samples, consisting of approximately 400 grams each from each of the following groups:

I. Bottom Oriented	II. Near Bottom Distribution	III. Anadromous	IV. Coastal Migratory
Winter flounder	Tautog	Alewife	Bluefish
Yellowtail founder	Cunner	Rainbow smelt	Atlantic herring
	Atlantic cod	Striped bass	Atlantic menhaden
	Pollock		Atlantic mackerel
	Hakes		

6 Mussel samples from four locations (immediate vicinity of discharge outfall, Manomet Point, Plymouth or Kingston Harbor, and Green Harbor in Marshfield) will be analyzed quarterly as follows:

One kilogram wet weight of mussel bodies, including fluid within shells will be collected. Bodies will be reduced in volume by drying at about 100°C. Sample will be compacted and analyzed by Ge(Li) gamma spectrometry or alternate technique, if necessary, to achieve a sensitivity of 5 pCi/kg for Cs-134, Cs-137, Co-60, Zn-65, and Zr-95; and 15 pCi/kg for Ce-144. Sensitivity values are to be determined in accordance with a 95% confidence level on k_a and a 50% confidence level on k_b (See HASL-300 for definitions).

The mussel shell sample from one location will be analyzed each quarter. One additional mussel shell sample will be analyzed semiannually. Unscrubbed shells to be analyzed will be dried, processed, and analyzed similarly to the mussel bodies.

Because of the small volume reduction in pre-processing of shells, sensitivities attained will be less than that for mussel bodies. The equipment and counting times to be employed for analyses of shells will be the same or comparable to that employed for mussel bodies so that the reduction in sensitivities (relative to those for mussel bodies) will be strictly limited to the effects of poorer geometry related to lower sample volume reduction. Shell samples not scheduled for analysis will be reserved (unscrubbed) for possible later analysis.

If radiocesium (Cs-134 and Cs-137) activity exceeds 200 pCi/kg (wet) in mussel bodies, these samples will be analyzed by radiochemical separation, electrodeposition, and alpha spectrometry for radioisotopes of plutonium, with a sensitivity of 0.4 pCi/kg.

7 Sediment samples from four locations (Manomet Point, Rocky Point, Plymouth Harbor, and head of Duxbury Bay) will be analyzed once per year (preferably early summer) as follows:

Cores will be taken to depths of 30-cm, minimum depth, wherever sediment conditions permit, by a hand-coring sampling device. If sediment conditions do not permit 30-cm deep cores, the deepest cores achievable with a hand-coring device will be taken. In any case, core depths will not be less than 14-cm. Core samples will be sectioned into 2-cm increments; surface and alternate increments will be analyzed, all others will be reserved. Sediment sample volumes (determined by core diameter and/or number of individual cores taken from any single location) and the counting technique will be sufficient to achieve sensitivities of 50 pCi/kg dry sediment for Cs-134, Cs-137, Co-60, Zn-65, and Zr-95 and 150 pCi/kg for Ce-144. In any case, individual core diameters will not be less than 2 inches.

The top 2-cm section from each core will be analyzed for Pu isotopes (Pu-238, Pu-239, and Pu-240) using radiochemical separations, electrodeposition, and alpha spectrometry with target sensitivity of 25 pCi/kg dry sediment. Two additional core slices per year (mid-depth slice from two core samples) will be similarly analyzed.

8 These locations may be altered in accordance with results of surveys discussed in Specification 8.1.B.

9 Minimum sensitivities for gamma exposure measurements are as follows:

Gamma exposure - 1 R/hr average exposure rate.
Gamma exposure survey - 1 R/hr exposure rate.

TABLE 8.1-2

AIR PARTICULATES, GASEOUS RADIOIODINE, AND SOIL SURVEILLANCE STATIONS

Sampling Location (Sample Designation)	Distance and Direction from Reactor
Offsite Stations	
East Weymouth (EW) (Control Station)	21 miles NW
Plymouth Center (PC)	4.0 miles W-WNW
Manomet Substation (MS)	2.5 miles SE
Cleft Rock Area (CR)	0.9 miles S
Onsite Stations	
Rocky Hill Road (ER)	0.8 miles SE
Rocky Hill Road (WR)	0.3 miles W-WNW
Overlook Area (OA)	0.03 miles W
Property Line (PL)	0 34 miles NW
Pedestrian Bridge (PB)	0.14 miles N
East Breakwater (EB)	0.35 miles ESE
Warehouse (WS)	0.03 miles SSE

TABLE 8.1-3

EXTERNAL GAMMA EXPOSURE SURVEILLANCE STATIONS[1]

Dosimeter Location (Designation)	Distance and Direction from Station
Onsite Stations	
Property Line (D)	0.17 miles NNW
Property Line (F)	0.12 miles NW
Property Line (I)	0.14 miles W
Property Line (G)	0.20 miles WSW
Rocky Hill Road (A)	0.12 miles SW
Property Line (H)	0.21 miles SSW
Public Parking Area (PA)	0.07 miles N-NNE
Pedestrian Bridge (PB)	0.1 miles NE
Overlook Area (OA)	0.03 miles W
East Breakwater (EB)	0.26 miles ESE
Property Line (C)	3.3 miles ESE-SE
Property Line (HB)	0.34 miles SE
Rocky Hill Road (B)	0.26 miles SSE
Microwave Tower (MT)	0.38 miles S
Emerson Road (EM)	0.68 miles SE-SSE
White Horse Road (WH)	0.89 miles SE-SSE
Property Line (E)	0.75 miles SSE-S
Rocky Hill Road (WR)	0.3 miles W-WNW
Property Line (J)	1.36 miles SSE-S
Property Line (K)	1.42 miles SSE-S
Rocky Hill Road (ER)	0.8 miles SE
Property Line (L)	0.40 miles E

TABLE 8.1-3 (Cont)

EXTERNAL GAMMA EXPOSURE SURVEILLANCE STATIONS[1]

Dosimeter Location (Designation)	Distance and Direction from Station
Onsite Stations (Cont)	
Warehouse (WS)	0.1 miles SE
Property Line (PL)	0.3 miles W
Offsite Stations	
Duxbury (SS)	6.25 miles SSW-SW
Kingston (KS)	10 miles WNW
North Plymouth (NP)	5.5 miles WNW
Plymouth Center (PC)	4.0 miles W-WNW
South Plymouth (SP)	3 miles WSW
Bayshore Drive (BD)	0.7 miles W-WNW
Cleft Rock Area (CR)	0.9 miles S
Manomet (MP)	2.25 miles ESE-S
Manomet (ME)	2.5 miles SE
Manomet (MS)	2.5 miles SSE
Manomet (MB)	3.5 miles SE-SSE
College Pond (CP)	6.5 miles SSW-SW
Sagamore (CS)	10 miles SSE-S
Plymouth Airport (SA)	8 miles WSW
East Weymouth (EW)[2]	21 miles NW
Saquish Neck (SN)[3]	4.6 miles NNW

1 Thermal Luminescent Dosimeters (TLDs)
2 Control Station
3 TLDs for this location will be provided to a third party and will be analyzed for gamma exposure whenever returned to Boston Edison Company.

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TABLE 8.1-4

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a]

Analysis	Water (pCi/kg)	Airborne Particulate or Gases (pCi/M^3)	Wet Solids (pCi/kg, wet)	Milk (pCi/g)	Food Products (pCi/kg, wet)	Dry Solids (pCi/kg, dry)
gross beta	4[b]	1×10^{-2}				
^{3}H	2000[d]					
^{54}Hn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			50
^{65}Zn	30		260			50
^{95}Zr	15					50
^{131}I	1	7×10^{-2}		1	60[c]	
$^{134,137}Cs$	15, 18	1×10^{-2}	130	15	60	50
^{140}Ba	15			15		
^{144}Ce						150

[a] Refer to ODCM for LLD definition.

[b] LLD for surface water.

[c] LLD for leafy vegetables.

[d] If no drinking water path exists, a value of 3000 pci/l may be used.

BASES:

7/8.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

7/8.1 MONITORING PROGRAM

A. Environmental Monitoring

An environmental radiological monitoring program is conducted to verify the adequacy of in-plant controls on the release of radioactive materials. The program is designed to detect radioactivity concentrations to ensure that radiation doses to individuals do not exceed the levels set forth in 10 CFR 50, Appendix I.

A supplemental monitoring program for sediments and mussels has been incorporated into the basic program (see Notes 6 and 7 to Table 8.1-1) as a result of an agreement with the Massachusetts Wildlife Federation. This supplemental program is designed to provide information on radioactivity levels at substantially higher sensitivity levels in selected samples to verify the adequacy (or, alternatively, to provide a basis for later modifications) of the long-term marine sampling schedules. As part of the supplemental program, analysis of mussels for isotopes of plutonium will be performed if radiocesium activity should exceed 200 pCi/kg in the edible portions.

The 200 pCi/kg radiocesium "action level" is based on calculations which show that if radiocesium from plant releases reached this level, plutonium could possibly appear at levels of potential interest.[1] The calculations also show that the dose delivered from these levels of plutonium would not be a significant portion of the total dose attributable to liquid effluents.

The program was also designed to be consistent, wherever applicable, with NUREG 0473.

Groundwater flow at the plant site is into Cape Cod Bay; therefore, terrestrial monitoring of groundwater is not included in this program.

Detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLD). The LLD in Table 8.1.4 is considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), curie, L.A.; "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry", Anal. Chem. 40, 586-93 (1968); and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

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[1] In measurable quantities having a potential dose (human food chain) significance comparable to other nuclides if present at their detection limits.

BASES:

7/8.1 MONITORING PROGRAM

B. Land Use Census

This section is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey, or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of 10CFR50, Appendix I, Section IV.B.3. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored, since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: 1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/m^2.

7/8.2 DOSE - LIQUID

This section is provided to implement the requirements of Sections II.A, III.A, and IV.A of 10CFR50, Appendix I, to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Because Pilgrim is not a site where plant operations can conceivably affect drinking water, none of these requirements are intended to assure compliance with 40 CFR 141. The dose calculations in the ODCM implement the requirements of 10CFR50, Appendix I, Section III.A to ensure that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

BASES:

7/8.3 DOSE - NOBLE GASES

This section is provided to implement the requirements of 10CFR50, Appendix I, Sections II.B, III.A, and IV.A to ensure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The surveillance requirements implement the requirements of 10CFR50, Appendix I, Section III.A to ensure that the actual exposure of a member of the public through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the site boundary will be based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

7/8.4 DOSE - IODINE-131, IODINE-133, RADIOACTIVE MATERIAL IN PARTICULATE FORM, AND TRITIUM

This section is provided to implement the requirements of Sections II.C, III.A and IV.A of 10 CFR50, Appendix I, to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements of 10CFR50, Appendix I, Section III.A to ensure that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods approved by the NRC for calculating the doses due to the actual release rates of the subject materials are required to be consistent with the methodology provided in Regulatory Guides 1.109 and 1.111. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, radioactive material in particulate form with half-lives greater than 8 days, and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in areas at and beyond the site boundary. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

BASES:

7/8.5 TOTAL DOSE

This section is provided to meet the dose limitations of 40CFR190 that have now been incorporated into 10CFR20 by 46 FR 18525. The specification requires the preparation and submittal of a special report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of 10CFR50, Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40CFR190 if the individual reactors remain within the reporting requirement level. The special report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40CFR190 limits. For the purposes of the special report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, except dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the limits of 40CFR190, a request for a variance in a special report in accordance with 40CFR190.11 and 10CFR20.405C is considered to be a timely request and fulfills the requirements of 40CFR190 until NRC staff action is completed. This is provided that the release conditions resulting in violation of 40CFR190 have not already been corrected. The variance only relates to the limits of 40CFR190, and does not apply in any way to the other requirements for dose limitation of 10CFR20. An individual is not considered a member of the public during any period in which he/she is engaged in any operation that is part of the nuclear fuel cycle.