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ACCESSION NBR:8505010131 DOC.DATE: 85/04/23 NOTARIZED: NO DOCKET #
FACIL:50-400 Shearon Harris Nuclear Power Plant, Unit 1, Carolina 05000400
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DENTON,H.R. Office of Nuclear Reactor Regulation, Director
see Report

SUBJECT: Forwards draft Rev 5 to NUREG-0452, "STS for Westinghouse PWRs," incorporating radiological effluent & ETS into updated draft version of Tech Specs.NRC comments expected in June 1985.

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Carolina Power & Light Company

SERIAL: NLS-85-112

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Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
UNIT NO. 1 - DOCKET NO. 50-400
TECHNICAL SPECIFICATIONS

REFERENCES: 1) Letter dated July 30, 1984 from A. B. Cutter (CP&L) to H. R. Denton (NRC)

2) Letter dated August 31, 1984 from S. R. Zimmerman (CP&L) to H. R. Denton (NRC)

Dear Mr. Denton:

Carolina Power & Light Company (CP&L) provides an updated copy of the "pen and ink" version of the SHNPP Technical Specifications (Attachment 1). This submittal supersedes our previous submittal made via Reference 1. This version of the SHNPP Technical Specifications is based upon Draft Revision 5 to NUREG-0452, "Standard Technical Specifications (STS) for Westinghouse Pressurized Reactors." The Radiological Effluent Technical Specifications are incorporated into the Technical Specifications. The Environmental Technical Specifications are included as Appendix B to the Technical Specifications. The Offsite Dose Calculation Manual was submitted via Reference 2. The Process Control Program will be submitted six months prior to fuel load.

This submittal contains the best information available to CP&L at this time. Comments from various industry groups on Draft Revision 5 of the STS have not yet been provided, but when they are available we may wish to make additional submittals consistent with those comments. Additional changes may be forthcoming as a result of the Westinghouse Owners' Group Technical Specification Optimization Programs (TOPS) and additional investigation of slave relay testing concerns.

Our original submittal provided a cross-reference between the Technical Specifications and certain requirements of the FSAR and SER (NUREG-1038) as well as justification for many CP&L proposed changes. A revised version of this cross-reference will be provided to you by May 31, 1985.

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Based upon the Standard Technical Specification review schedule (Attachment 2), CP&L is expecting NRC comments in June 1985. Please note that Attachment 2 indicates a date of June 1, 1986 for a full power license. However, as discussed at our March 22, 1985 pre-application meeting, CP&L expects to have the necessary activities complete to allow the NRC to issue a full power license on March 1, 1986.

If you have any questions, please contact Mr. Gregg A. Sinders at (919) 836-8168.

Yours very truly,

A. B. Cutter - Vice President
Nuclear Engineering & Licensing

GAS/mf (672GAS)
Enclosures

cc: Mr. B. C. Buckley (NRC)
Mr. G. F. Maxwell (NRC-SHNPP)
Mr. J. P. O'Reilly (NRC-RII)
Mr. C. Moon (NRC-SSPB)
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Mr. G. O. Bright (ASLB)
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Attachment 1

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NUREG-0452
Revision 5

Standard Technical Specifications for Westinghouse Pressurized Water Reactors

Revision Issued
Supersedes NUREG-0452, Revision 4

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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UNITED STATES NUCLEAR REGULATORY COMMISSION

NUREG-0452, REVISION 5 STANDARD TECHNICAL SPECIFICATIONS FOR WESTINGHOUSE PRESSURIZED WATER REACTORS

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FOREWORD

The following paragraphs briefly describe the applicability, format, and implementation of the Westinghouse Standard Technical Specification package.

APPLICABILITY

This Standard Technical Specification (STS) has been structured for the broadest possible use on Westinghouse plants currently being reviewed for an Operating License. Accordingly, the document contains specifications applicable to plants with (1) either 3 or 4 loops and (2) with and without loop stop valves. In addition, four separate and discrete containment specification sections are provided for each of the following containment types: Atmospheric, Ice Condenser, Subatmospheric, and Dual. Optional specifications are provided for those features and systems which may be included in individual plant designs but are not generic in their scope of application. Alternate specifications are provided in a limited number of cases to cover situations where alternate specification requirements are necessary on a generic basis because of design differences.

FORMAT

The format of the STS addresses the categories required by 10 CFR and consists of six sections covering the areas of: Definitions, Safety Limits and Limiting Safety System Settings, Limiting Conditions for Operation, Surveillance Requirements, Design Features, and Administrative Controls. The Limiting Conditions for Operation and Surveillance Requirements (Sections 3 and 4) are presented in a combined format with each LCO appearing at the top of the page followed immediately by the applicable Surveillance Requirements. The combined Section 3/4 is further subdivided into twelve subsections covering the areas of:

1. Reactivity Control,
2. Power Distribution,
3. Instrumentation,
4. Reactor Coolant System,
5. Emergency Core Cooling System,
6. Containment Systems,
7. Plant Systems,
8. Electrical,
9. Refueling Operations,
10. Special Test Exceptions,
11. Radioactive Effluents, and
12. Radiological Environmental Monitoring.

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The values of those parameters and variables which may vary because of plant design appear as either blanks or bracketed numbers throughout the STS. The actual value for each parameter will be provided by individual applicants as appropriate for their plants. The values in brackets are for illustration only.

ANNOTATIONS

Although this volume of Standard Technical Specifications is republished periodically (approximately annually) changes are made to sections of it as the NRC deems appropriate. To assist the user of this document, certain annotations have been provided, as follows:

1. The date at the bottom of each page is the date of the last change made to that page.

2. The vertical striping in the margin of each page indicates the location of changes made on that page since publication of Revision 4 of this volume in Fall 1981.

IMPLEMENTATION

The implementation of the STS on an individual license application will proceed in five phases. The major steps within each phase are indicated below.

Phase I (at least 12 months prior to scheduled licensing (fuel load date) of facility).

The applicant should:

1. Obtain copies of the appropriate STS for his facility from the NRC licensing project manager.

2. Select the appropriate containment specification section and set aside the non-applicable containment sections and related bases.

3. Identify and mark those specifications not required because of plant design or other factors. Specifications within this category should be retained in position within the document package for later review and discussion.

4. Identify those areas where specifications are required but are not provided in the STS (should be related to the facility design and NRC staff requirements stated in Safety Evaluation Report).

5. Provide the applicable values of the parameters and variables identified by blanks or brackets in the STS. This information must be consistent with the SAR and other supporting documents.

6. Provide the figures, graphs, and other information required to complete the STS document package.

7. Provide written justification for any changes to STS requirements including plant-specific and site-specific features. This discussion should include bases for change, references, supporting information, and a marked up STS page(s) with the proposed changes.

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Phase II (at least 6 months prior to scheduled licensing (fuel load date) of facility, if possible)

1. The Commission staff (Technical Specification project manager) will review the information provided in the marked up STS document package resulting from the Phase I preparation.

2. The Technical Specification project manager will prepare a draft TS for the applicants' facility which includes all changes to the STS that are acceptable to the NRC staff without further justification. The applicant will be provided a copy of this draft TS for his review.

3. Applicant/NRC staff meeting will be held to resolve noted differences of position and other related comments from the applicant, vendor, and A.E. on the draft TS. Issues requiring resolution during Phase III will be identified.

Phase III (at least 4 months prior to scheduled licensing (fuel load date) of facility, if possible)

1. The Commission will provide a Proof and Review edition of the Technical Specification for final review by all parties (including NRC technical review branches, applicable Regional Office, and applicant) based upon the resolution of comments and positions in Phase II.

2. Final comments and corrections will be incorporated into the document as approved by the Commission staff.

3. Issues, if any, requiring the Appeal Process by the applicant will be identified.

Phase IV (1 month prior to scheduled licensing (fuel load date) of facility)

1. The Commission will provide a Final Draft edition of the Technical Specification for final review and certification by the applicant to accurately reflect the plant as-built, FSAR as supplemented, and SER.

2. Final comments, corrections and resolution of any outstanding issues will be incorporated into the document as certified by the applicant.

Phase V

The Technical Specifications will be issued by the Commission as Appendix "A" to the Operating License prior to fuel loading.

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SECTION 1.0

DEFINITIONS

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1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

~~ANALOG~~ CHANNEL OPERATIONAL TEST

1.~~3~~6 An ~~ANALOG~~ CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ~~ANALOG~~ CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.~~4~~3 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.~~5~~4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.~~6~~5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

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DEFINITIONS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in ~~Table [3.6-1] of~~ Specification ~~[3.6.3]~~, 3/4.6.3

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification ~~[~~3.6.1.3~~]~~,

d. The containment leakage rates are within the limits of Specification ~~[~~3.6.1.2~~]~~, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~[~~Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." ~~or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977]~~.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample. An analysis for $\bar{E}$ shall consist of a quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is ~~def~~ identified in the (cont. on next page)

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radionuclides shall be used in the determination of $\bar{E}$ of the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identified with a 95% confidence level.

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ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

EXCLUSION AREA BOUNDARY

1.13 THE EXCLUSION AREA BOUNDARY SHALL BE THAT LINE BEYOND WHICH THE LAND IS NOT CONTROLLED TO LIMIT ACCESS BY THE LICENSEE.

FREQUENCY NOTATION

1.~~13~~ 14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.15 A GASEOUS RADWASTE TREATMENT SYSTEM IS ANY SYSTEM DESIGNED AND INSTALLED TO REDUCE RADIOACTIVE GASEOUS EFFLUENTS BY COLLECTING PRIMARY COOLANT SYSTEM OFF-GASES FROM THE PRIMARY SYSTEM AND PROVIDING FOR DELAY OR HOLDUP FOR THE PURPOSE OF REDUCING THE TOTAL RADIOACTIVITY PRIOR TO RELEASE TO THE ENVIRONMENT.

IDENTIFIED LEAKAGE

1.~~14~~ 16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.~~15~~ 17 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.~~16~~ 18 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 licensee, 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 plant.

OFFSITE DOSE CALCULATION MANUAL

1.~~17~~ 19 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

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DEFINITIONS

OPERABLE - OPERABILITY

1.~~18~~ 20 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, 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).

OPERATIONAL MODE - MODE

1.~~19~~ 21 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.~~20~~ 22 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter {14.0} of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.~~21~~ 23 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.~~22~~ 24 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

PURGE - PURGING

1.~~23~~ 25 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

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QUADRANT POWER TILT RATIO

1.2~~4~~6 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.2~~5~~7 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2775 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.2~~6~~8 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.2~~7~~9 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

~~SHIELD BUILDING INTEGRITY~~

~~1.28 SHIELD BUILDING INTEGRITY shall exist when:~~

~~a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed,~~

~~b. The Shield Building Filtration System is OPERABLE, and~~

~~c. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.~~

SHUTDOWN MARGIN

1.~~29~~ 30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all ~~full length~~ rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.3~~0~~1 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

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SLAVE RELAY TEST

1.3~~1~~2 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.3~~2~~3 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.3~~3~~4 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.3~~4~~5 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, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.3~~5~~6 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.3~~6~~7 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.3~~7~~8 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

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UNRESTRICTED AREA

1.3~~8~~9 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.~~39~~40 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.4~~0~~1 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

~~WASTE GAS HOLDUP SYSTEM~~

~~1.41 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

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TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

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TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in ~~Figures~~ Figure 2.1-1 ~~and 2.1-2 for n and n-1~~ FOR 3 loop operation, ~~respectively.~~

APPLICABILITY: MODES 1 and 2.

ACTION:

a. Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

b. OPERATION WITH ONLY TWO LOOPS OPERATING BELOW THE P-8 INTERLOCK AND OPERATION WITH NO LOOPS OPERATING BELOW THE P-7 INTERLOCK ARE GOVERNED BY SPECIFICATION 3.4.1.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig/ EXCEPT DURING HYDROSTATIC TESTING.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

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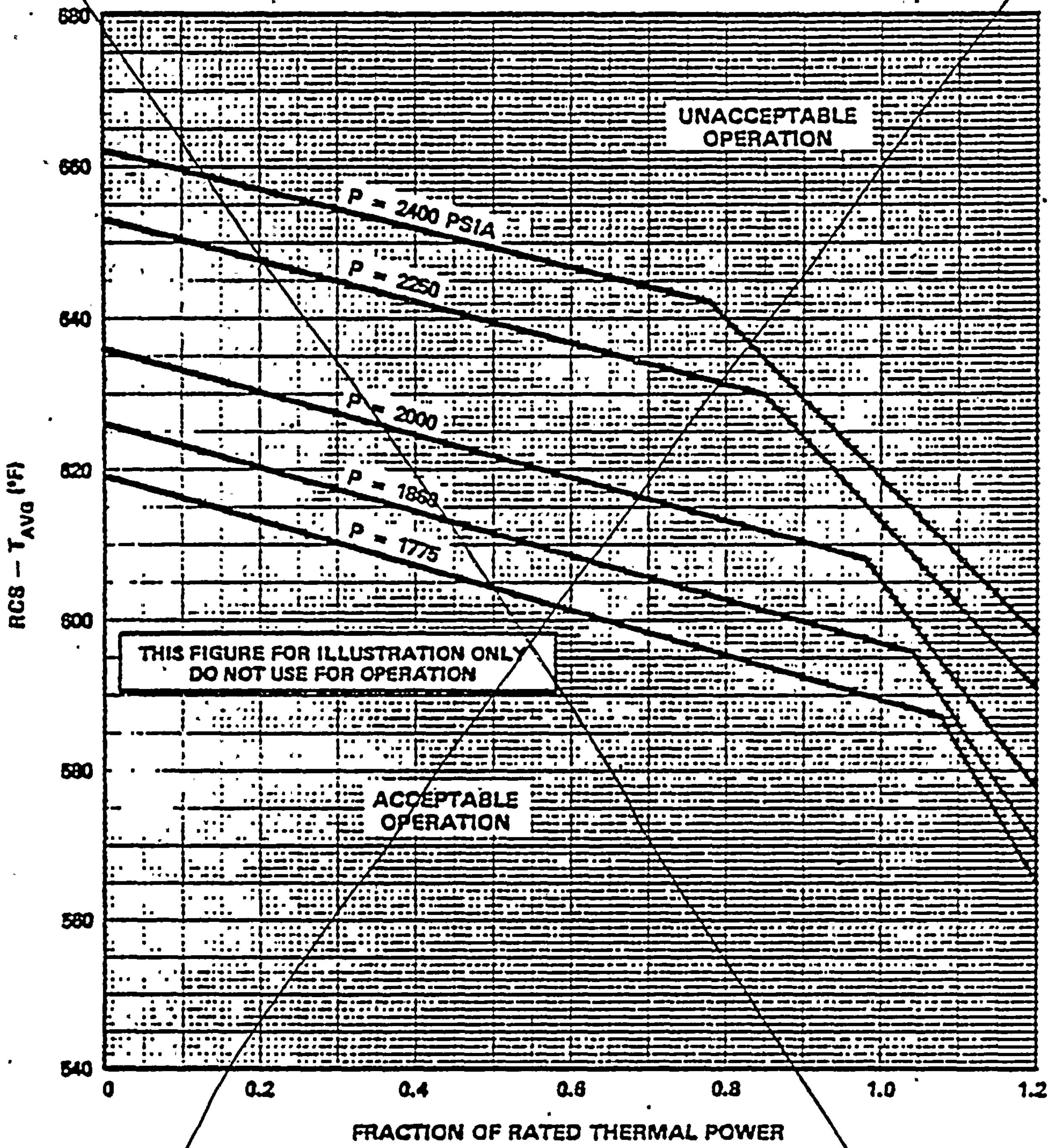

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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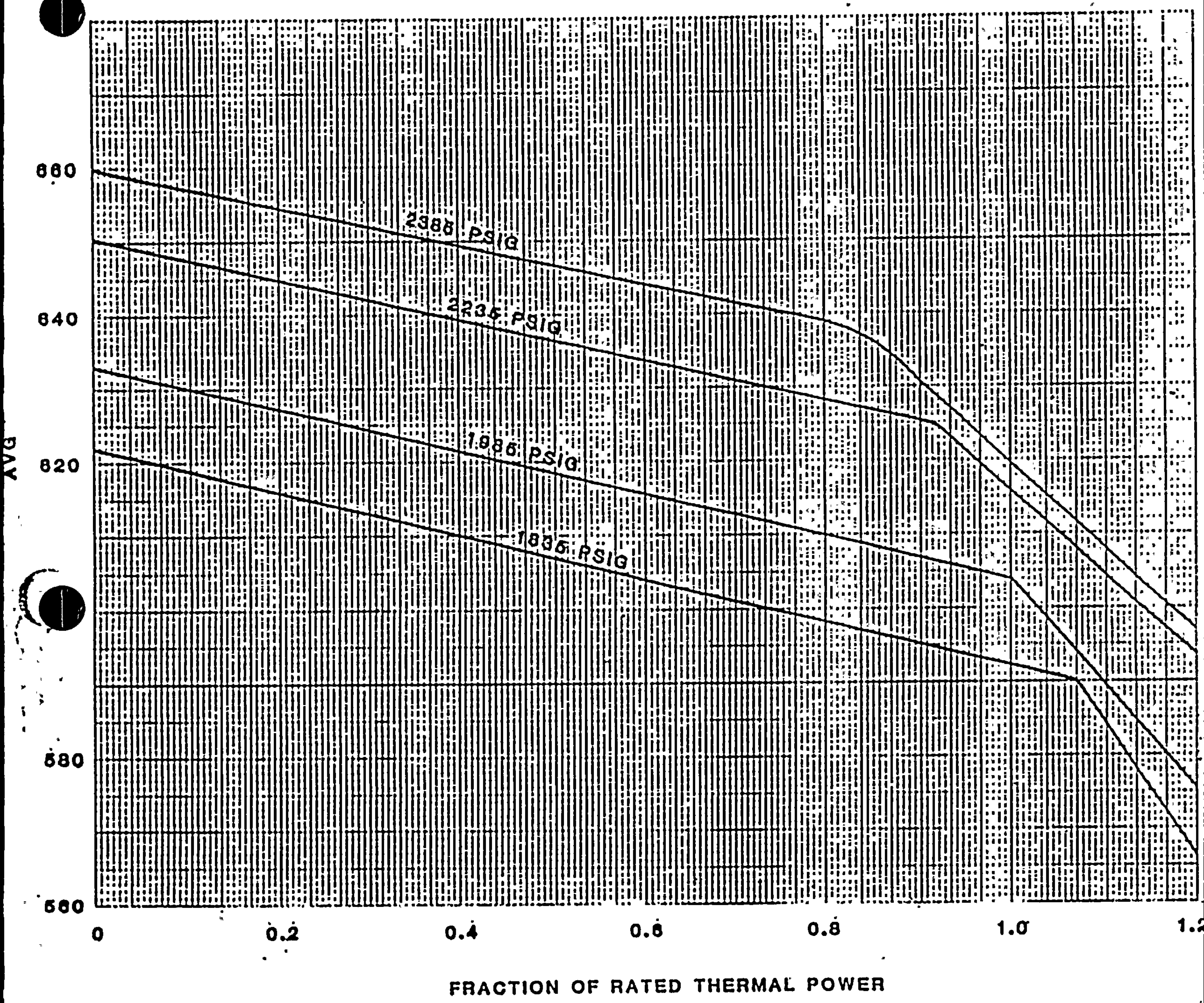

FIGURE 2.1-1

REACTOR CORE SAFETY LIMITS

THREE LOOPS IN OPERATION

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2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, ~~either:~~ PLACE THE CHANNEL IN THE TRIPPED CONDITION WITHIN 1 HOUR, AND WITHIN THE FOLLOWING 12 HOURS EITHER:

1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine ~~within 12 hours~~ that Equation 2.2-1 was satisfied for the affected channel, or

2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $\quad Z + R + S \leq TA$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

c. WITH A ~~N~~ ~~ESFAS~~ REACTOR TRIP SYSTEM INSTRUMENTATION CHANNEL OR INTERLOCK INOPERABLE, TAKE THE ACTION SHOWN IN TABLE 3.3-1

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux a. High Setpoint	[7.5]	~~[4.56]~~ 6	0	≤[109]% of RTP**	≤[111.~~2~~ 1]% of RTP**
b. Low Setpoint	[8.3]	~~[4.56]~~ 6	0	≤[25]% of RTP**	≤[27.~~2~~ 1]% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	~~[2.0]~~ 1.6	[0.5]	0	≤[5]% of RTP** with a time constant ≥[2] seconds	≤[6.~~8~~ 3]% of RTP** with a time constant ≥[2] seconds
4. Power Range, Neutron Flux, High Negative Rate	~~[2.0]~~ 1.6	[0.5]	0	≤[5]% of RTP** with a time constant ≥[2] seconds	≤[6.~~8~~ 3]% of RTP** with a time constant ≥[2] seconds
5. Intermediate Range, Neutron Flux	[17.0]	[8.41]	0	≤[25]% of RTP**	≤~~[31]~~ 30.9% of RTP**
6. Source Range, Neutron Flux	[17.0]	[10.01]	0	≤[10^5] cps	≤[1.4 x 10^5] cps
7. Overtemperature ΔT	~~[6.7]~~ 7.1	~~[2.79]~~ 4.7	~~[0.8]~~ 2.0	See Note 1	See Note 2
8. Overpower ΔT	[4.~~3~~] 6	[1.~~3~~] 4	~~[0.2]~~ 0.1	See Note 3	See Note 4
9. Pressurizer Pressure-Low	~~[6.8]~~ 5.0	~~[0.71]~~ 2.2	[1.5]	≥~~[1900]~~ 1960 psig	≥~~[1886]~~ 1946 psig
10. Pressurizer Pressure-High	~~[3.1]~~ 7.5	~~[0.71]~~ 5.0	[1.5]	≤[2385] psig	≤[239~~6~~ 9] psig
11. Pressurizer Water Level-High	~~[5.0]~~ 8.0	[2.~~18~~] 2	[1.5]	≤[92]% of instrument span	≤[93.8]% of instrument span
12. Loss of ~~Reactor Coolant~~ Flow~~-Low~~	[2.5]	[1.~~0~~] 9	~~[1.6]~~ 0.7	≥[90]% of loop design flow*	≥[89.~~2~~ 4]% of loop design flow*

*Loop design flow = ~~[95,700]~~ 97,600 gpm

**RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low	~~[30.0]~~ 19.2	~~[27.18]~~ 18.2	[1.5]	≥~~[32.3]~~ 38.3% of narrow range instrument span	≥~~[30.4]~~ 37.8% of narrow range instrument span
14. Steam/Feedwater Flow Mismatch ~~Coincident With~~	~~[16.0]~~ 20.0	~~[13.24]~~ 4.6	~~[1.5]~~ 3.2	≤[40]% of full steam flow at RTP**	≤~~[42.5]~~ 43.1% of full steam flow at RTP**
Steam Generator Water Level-Low-~~Low~~ COINCIDENT WITH	~~[12.0]~~ 19.2	~~[9.18]~~ 6.70	[1.5]	≤~~[32.3]~~ 38.3% of narrow range instrument span	≤~~[30.4]~~ 36.6% of narrow range instrument span
15. Undervoltage - Reactor Coolant Pumps	~~[2.3]~~ 44.7	~~[1.28]~~ 1.3	0.0	≥~~[4830]~~ LATER volts	≥~~[4760]~~ LATER volts
16. Underfrequency - Reactor Coolant Pumps	~~[7.5]~~ 5.0	3.0	~~[0.1]~~ 0	≥~~[57.5]~~ 57.5 Hz	≥~~[57.1]~~ 57.3 Hz
17. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥~~[800]~~ LATER psig	≥~~[750]~~ LATER psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥~~[1]~~ LATER % open	≥~~[1]~~ LATER % open
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

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~~W-STS~~ SHEARON HARRIS UNIT 1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	≥[1×10^{-10}] amp	≥[6×10^{-11}] amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤[10]% of RTP**	≤[12.~~2~~ 1]% of RTP**
2) P-13 input	N.A.	N.A.	N.A.	≤[10]% RTP** Turbine Impulse Pressure Equivalent	≤[12.~~2~~ 1]% RTP** Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤~~[48]~~ 49% of RTP**	≤~~[50.2]~~ 51.1% of RTP**
~~d. Power Range Neutron Flux, P-9~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≤[50]% of RTP**~~	~~≤[52.2]% of RTP**~~
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	≥[10]% of RTP**	≥[7.~~8~~ 9]% of RTP**
f. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	≤[10]% RTP** Turbine Impulse Pressure Equivalent	≤[12.~~2~~ 1]% RTP** Turbine Impulse Pressure Equivalent
20. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \left(\frac{1}{1 + \tau_6 S}\right) - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = [8] s, τ_2 = [3] s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = ~~[2]~~ 0 s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	~~[1.09]~~; 1.10
K_2	=	~~[0.0130]~~/°F; 0.0182
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = [33] s, τ_5 = [4] s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = ~~[2]~~ 0 s;

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	[588.~~5]~~ 8 °F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	~~[0.000671]~~ 0.000828/psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between ~~-[35]~~ 34% and + ~~[7]~~ 9.0%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

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(2) For each percent that the magnitude of $q_t - q_b$ exceeds ~~-[35]~~ 34%, the ΔT Trip Setpoint shall be automatically reduced by ~~[1.26]~~ 2.02% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + ~~[7]~~ 9%, the ΔT Trip Setpoint shall be automatically reduced by ~~[1.05]~~ 1.83% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~[3.8]~~ 1.6%

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:	ΔT	=	As defined in Note 1,
	$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1,
	τ_1, τ_2	=	As defined in Note 1,
	$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1,
	τ_3	=	As defined in Note 1,
	ΔT_0	=	As defined in Note 1,
	K_4	=	~~[1.09]~~ 1.086,
	K_5	=	[0.02]/°F for increasing average temperature and 0 for decreasing average temperature,
	$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
	τ_7	=	Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = [10] s,
	$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
	τ_6	=	As defined in Note 1,

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00159 ~~[0.00128]~~/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ [588.~~5]~~ 8°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~[3.0]~~ 3.2%.

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

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NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

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2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the ACTUAL local heat flux and is indicative of the margin to DNB. CALCULATED

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure~~s~~ ~~[~~2.1-1~~] and [2.1-2]~~ shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

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SAFETY LIMITS

BASES

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor PRESSURE vessel, ~~pressurizer,~~ and the RCS piping~~, valves~~ and fittings are designed to Section III, DIVISION I of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% ~~(2735 psig)~~ of design pressure, WHICH IS EQUIVALENT TO 2735 psig. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

THE REACTOR COOLANT SYSTEM VALVES ARE DESIGNED TO SECTION III OF THE ASME CODE AND ARE PERMITTED A MAXIMUM TRANSIENT PRESSURE OF 120% OF COMPONENT DESIGN PRESSURE WHICH IS EQUIVALENT TO 2985 PSIG.

The entire RCS is hydrotested at 125% ~~(3110~~ 3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

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2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy. INSERT from NEXT PAGE.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for ~~REPORTABLE EVENTS.~~ determination of OPERABILITY.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

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INSERT FOR ~~SECTION~~ PAGE B 3/4 3-2

FOR EXAMPLE, IF A BISTABLE HAS A TRIP SETPOINT OF ≤ 100%, HAS A SPAN OF 125%, AND HAS A CALIBRATION ACCURACY OF ± .50% of SPAN, THEN THE BISTABLE IS CONSIDERED TO BE ADJUSTED TO THE TRIP SETPOINT AS LONG AS THE "AS MEASURED" VALUE FOR THE BISTABLE IS ≤ 100.62%.

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LIMITING SAFETY SYSTEM SETTINGS

BASES

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than ~~1.30~~. THE LIMITING VALUE

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Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active. NO CREDIT WAS TAKEN FOR OPERATION OF THE TRIPS ASSOCIATED WITH EITHER THE INTERMEDIATE OR SOURCE RANGE CHANNELS IN THE ACCIDENT ANALYSIS. HOWEVER, THEIR FUNCTIONAL CAPABILITY AT THE SPECIFIED TRIP SETTINGS IS REQUIRED BY THIS SPECIFICATION TO ENHANCE THE OVERALL RELIABILITY OF THE REACTOR PROTECTION SYSTEM.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Optional for Plants Permitted n-1 Loop Operation

Operation with a reactor coolant loop out of service below the (n) loop P-8 Setpoint does not require Reactor Trip System Setpoint modification because the P-8 Setpoint and associated trip will prevent DNB during (n-1) loop operation exclusive of the Overtemperature ΔT Setpoint. (n-1) loop operation above the (n) loop P-8 Setpoint is permissible after resetting the K1 input to the Overtemperature ΔT channels and raising the P-8 Setpoint to its (n-1) loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of

Overpower ΔT (Continued)

change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Loss of ~~Reactor Coolant~~ Flow

The Loss of ~~Low Reactor Coolant~~ Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on Loss of ~~low reactor coolant~~ flow in more than one loop and below P-7 the trip function is automatically blocked.

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~~Optional for Plants Permitted n-1 Loop Operation~~

~~The P-8 Setpoint trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when [n-1] loops are in operation and the Overtemperature ΔT Trip Setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT Trip Setpoint adjusted to the value specified for [n-1] loop operation, the P-8 trip at [76%] RATED THERMAL POWER will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients with [n-1] loops in operation.~~

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to ~~[1.42~~ 1.627 x 10^6] lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below ~~[25%]~~ ~~38%~~ 38.3%, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed [1.2] seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed [0.3] seconds.

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Undervoltage and Underfrequency - Reactor Coolant Pump Busses (Continued)

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-~~8~~ 7 (a power level of approximately ~~50%~~ 10% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-~~8~~. 7

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), ~~provides a backup block for Source Range Neutron Flux doubling,~~ and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage and underfrequency, Turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

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Reactor Trip System Interlocks (Continued)

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. ~~and one or more reactor coolant pump breakers open.~~ On decreasing power, the P-8 automatically blocks the above listed trips.

~~P-9 On increasing power, P-9 automatically enables Reactor trip on Turbine trip. On decreasing power, P-9 automatically blocks Reactor trip on Turbine trip.~~

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

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SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,

b. At least HOT SHUTDOWN within the following 6 hours, and

c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval; but

b. The combined time interval for any three consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i);

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SURVEILLANCE REQUIREMENTS (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities;

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements; and

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

Capitalize the Titles of the Following System and Component Names:

Boric Acid Tanks

Boric Acid Transfer Pump

Charging/Safety Injection Pump
(instead of charging pump)

Refueling Water Storage Tank (RWST)

Safety Injection Actuation

Digital Rod Position Indication System

Demand Position Indication System

Reactor Trip System

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~[1.6%] Δk/k~~ 1770 pcm for ~~[N]~~ 3 loop operation.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than ~~[1.6%] Δk/k~~ 1770 pcm, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~[1.6%] Δk/k~~ 1770 pcm:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

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SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3 or 4, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ~~± 1% Δk/k~~ ±1000 pcm at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification [4.1.1.1.1e.], above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

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SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to ~~1% Δk/k~~ 2000 pcm.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than ~~1% Δk/k~~ 2000 pcm, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1% Δk/k~~ 2000 pcm:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

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REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than ~~[0] Δk/k~~ pcm/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition; and

b. Less negative than ~~[3.9] x 10⁻⁴ Δk/k~~ −42 pcm/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only**.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 ~~Δk/k/°F~~ pcm/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

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SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to −33 pcm/°F ~~−[3.0] x 10^{-4} $\Delta k/k/°F$~~ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than ~~−[3.0] x 10^{-4} $\Delta k/k/°F$,~~ −33 pcm/°F the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

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MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to ~~[541]~~ 551°F.

APPLICABILITY: MODES 1 and 2* **.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551 ~~[541]~~°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to ~~[541]~~ 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than ~~[551]~~ 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

*With K_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

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REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System if the boric acid ~~storage~~ tank in Specification [3.1.2.5a.] is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification [3.1.2.5b.] is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

~~a. At least once per 7 days by verifying that the temperature [of the heat traced portion] of the flow path is greater than or equal to [65]°F when a flow path from the boric acid tanks is used, and~~

a ~~b~~. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging/SAFETY INJECTION pumps to the RCS.

APPLICABILITY: MODES 1, 2, 3, and 4.*

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least ~~1% Δk/k~~ 2000pcm at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

~~a. At least once per 7 days by verifying that the temperature [of the heat traced portion] of the flow path from the boric acid tanks is greater than or equal to [65]°F when it is a required water source;~~

a ~~b~~. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b ~~c~~. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a SAFETY INJECTION test signal; and

c ~~d~~. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

~~*Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to [275]°F.~~ *THE PROVISIONS OF SPECIFICATIONS 3.0.4 AND 4.0.4 ARE NOT APPLICABLE FOR ENTRY INTO MODE 4 WITH THE RCS COLD LEG TEMPERATURES GREATER THAN 250°F FOR THE CHARGING PUMP DECLARED INOPERABLE PURSUANT TO SPECIFICATION 4.1.2.3.2 PROVIDED THAT THE CHARGING PUMP IS RESTORED TO OPERABLE STATUS WITHIN 4 HOURS PRIOR TO THE TEMPERATURE OF ONE OR MORE OF THE RCS COLD LEGS EXCEEDING ~~350~~ 275°F

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REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification ~~[3.1.2.1]~~ shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

OR INSERVICE SUPPLYING FLOW TO THE REACTOR COOLANT SYSTEM AND REACTOR COOLANT PUMP SEALS.

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across the pump of greater than or equal to 2446 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

*AN INOPERABLE PUMP MAY BE ENERGIZED FOR TESTING ~~OR FOR FILLING ACCUMULATORS~~ PROVIDED THE DISCHARGE OF THE PUMP HAS BEEN ISOLATED FROM THE RCS BY A CLOSED ISOLATION VALVE WITH POWER REMOVED FROM THE VALVE OPERATOR, OR BY A MANUAL ISOLATION VALVE SECURED IN THE CLOSED POSITION

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REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two* charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

2000 pcm

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least ~~1% Δk/k~~ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

OR INSERVICE SUPPLYING FLOW TO THE REACTOR COOLANT SYSTEM AND REACTOR COOLANT PUMP SEALS

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across each pump of greater than or equal to 2446 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above allowed OPERABLE pump, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one or more of the Reactor Coolant System (RCS) cold legs is less than or equal to ~~[275]~~°F by verifying that the motor circuit breakers are secured in the open position. 250

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to ~~[275]~~°F.
250

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REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid ~~Storage System~~ TANK with:

1) A minimum contained borated water volume of ~~5222~~ 5400 gallons WHICH IS EQUIVALENT TO 13 % INDICATED LEVEL.

2) A minimum boron concentration of [7000] ppm, and

3) A minimum solution temperature of [65]°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of ~~58412~~ LATER gallons WHICH IS EQUIVALENT TO ~~[ ]~~ LATER % INDICATED LEVEL

2) A minimum boron concentration of [2000] ppm, and

3) A minimum solution temperature of ~~[35]~~ 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the boric acid ~~storage~~ tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the [outside] air temperature is less than ~~[35]~~ 40°F.

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid ~~Storage System~~ Tank with:

1) A minimum contained borated water volume of 16300 gallons, which is equivalent to 45% indicated level;

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of Later gallons,

2) A minimum boron concentration of 2000 ppm,

3) A minimum solution temperature of [~~25~~] 40°F, and

4) A maximum solution temperature of [~~100~~]°F. Later

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the Boric Acid ~~Storage System~~ TANK inoperable and being used as one of the above required borated water sources, restore the ~~system~~ boric acid tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least ~~1% Δk/k~~ 2000 pcm at 200°F; restore the Boric Acid ~~Storage System~~ TANK to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) Verifying the Boric Acid TANK ~~Storage System~~ solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the [outside] air temperature is ~~either~~ less than ~~[35]~~ 40°F ~~or greater than [100]°F~~.

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REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All ~~full-length~~ shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more ~~full-length~~ rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one ~~full-length~~ rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one ~~full-length~~ rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

1. The rod is restored to OPERABLE status within the above alignment requirements, or

2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure~~s~~ ~~[~~3.1-1~~]~~ ~~and [3.1-2]~~. The THERMAL POWER level shall be restricted pursuant to Specification ~~[~~3.1.3.6~~]~~ during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each ~~full-length~~ rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each ~~full-length~~ rod not fully inserted shall ~~in the core shall~~ be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

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TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE ~~FULL LENGTH~~ ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

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REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

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REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the ~~control~~ rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **.

ACTION:

With less than the above required position indicator(s) OPERABLE, ~~immediately open the Reactor Trip System breakers.~~ EITHER RESTORE THE INDICATOR TO OPERABLE WITHIN 8 HOURS OR OPEN THE REACTOR TRIP SYSTEM BREAKERS

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel at least once per 18 months.

*With the Reactor Trip System breakers in the closed position.

**See Special Test Exceptions Specification 3.10.5.

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ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual ~~full-length~~ ~~(~~shutdown and control~~)~~ rod 3.0 drop time from the fully withdrawn position shall be less than or equal to ~~[2.2]~~ seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to ~~[541]~~ 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any ~~full-length~~ rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with ~~N-1~~ two reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to ~~either:~~

~~1.~~ less than or equal to ~~[~~66~~]~~% of RATED THERMAL POWER. ~~when the reactor coolant stop valves in the nonoperating loop are open, or~~

~~2. Less than or equal to [76]% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.~~

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of ~~full-length~~ SHUTDOWN AND CONTROL rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

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REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification [4.1.3.1.2], within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification [3.1.3.1].

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.

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CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure~~s~~ [3.1-1] ~~and [3.1-2].~~

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification [4.1.3.1.2]:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figures, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the rod insertion limit monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.

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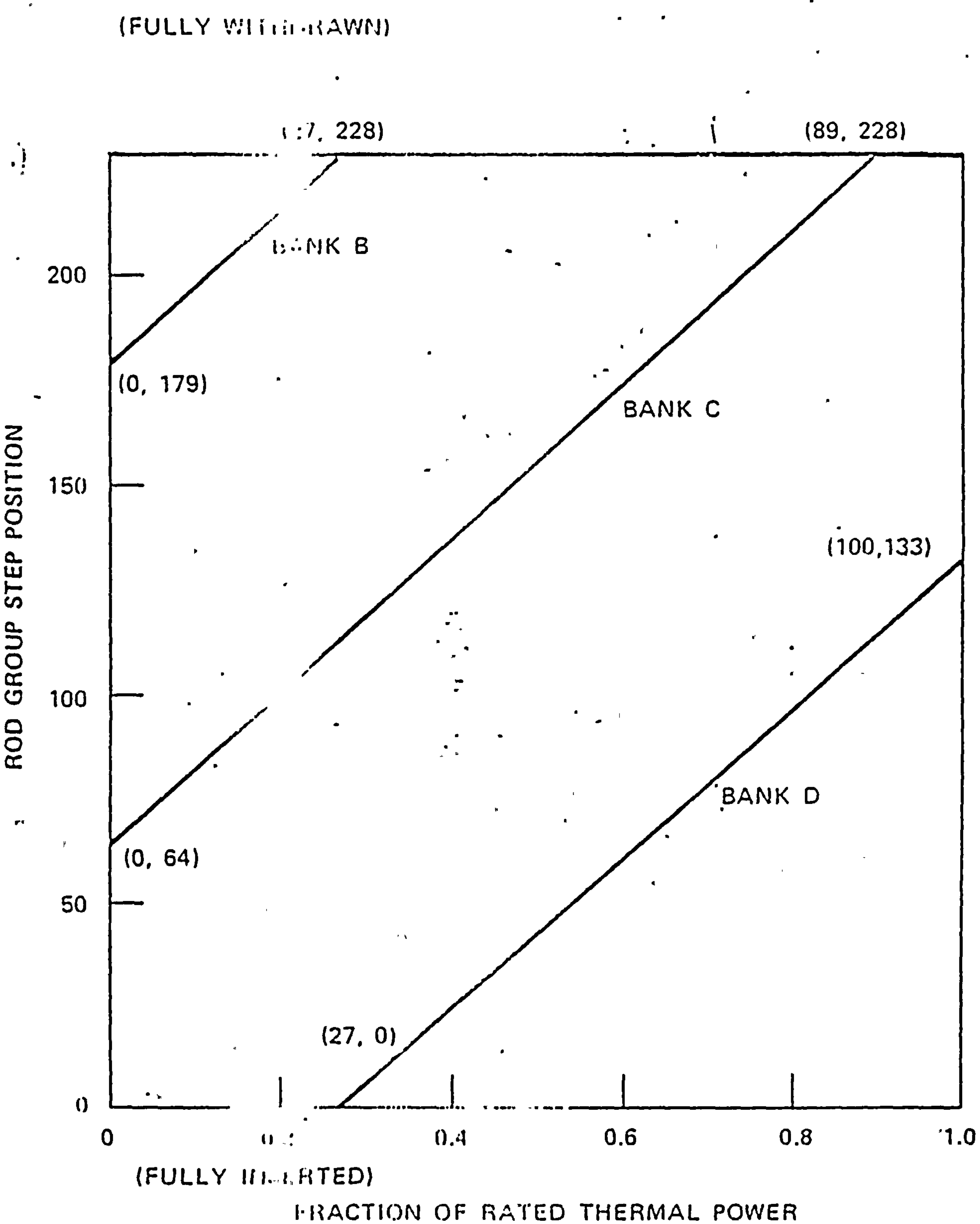

Figure 3.1-1. Rod Group Insertion Limits versus Thermal Power – Three Loop Operation

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~~FIGURE 3.1-2~~

~~ROD BANK INSERTION LIMITS VERSUS THERMAL POWER~~
~~THREE LOOP ØPERATION~~

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) ~~about~~ FROM the target ~~flux difference:~~ AFD VALUE;

a. ± 5% for core average accumulated burnup of less than or equal to 3000 MWD/MTU; and

b. + 3%, -12% for core average accumulated burnup of greater than 3000 MWD/MTU.

The indicated AFD may deviate outside the above required target band at greater than or equal to 50% but less than ~~90%~~ ~~81%~~ 90 of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to ~~90%~~ ~~81%~~ 90 of RATED THERMAL POWER, within 15 minutes either:

1. Restore the indicated AFD to within the target band limits, or

2. Reduce THERMAL POWER to less than ~~90%~~ 90 ~~81%~~ of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than ~~90%~~ ~~81%~~ 90 but equal to or greater than 50% of RATED THERMAL POWER, reduce:

1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

2. The Power Range Neutron Flux* ** - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

*See Special Test Exceptions Specification 3.10.2.

**Surveillance testing of the Power Range Neutron Flux Channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

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POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MAINTAIN

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours ~~and with~~ THERMAL POWER less than 50% ~~but greater than 15%~~ of RATED THERMAL POWER. The THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band AND THE INDICATED AFD OUTSIDE OF THE ABOVE REQUIRED BAND FOR LESS THAN 1 HOUR OF CUMMULATIVE PENALTY DEVIATION TIME DURING THE PREVIOUS 24 HOURS.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

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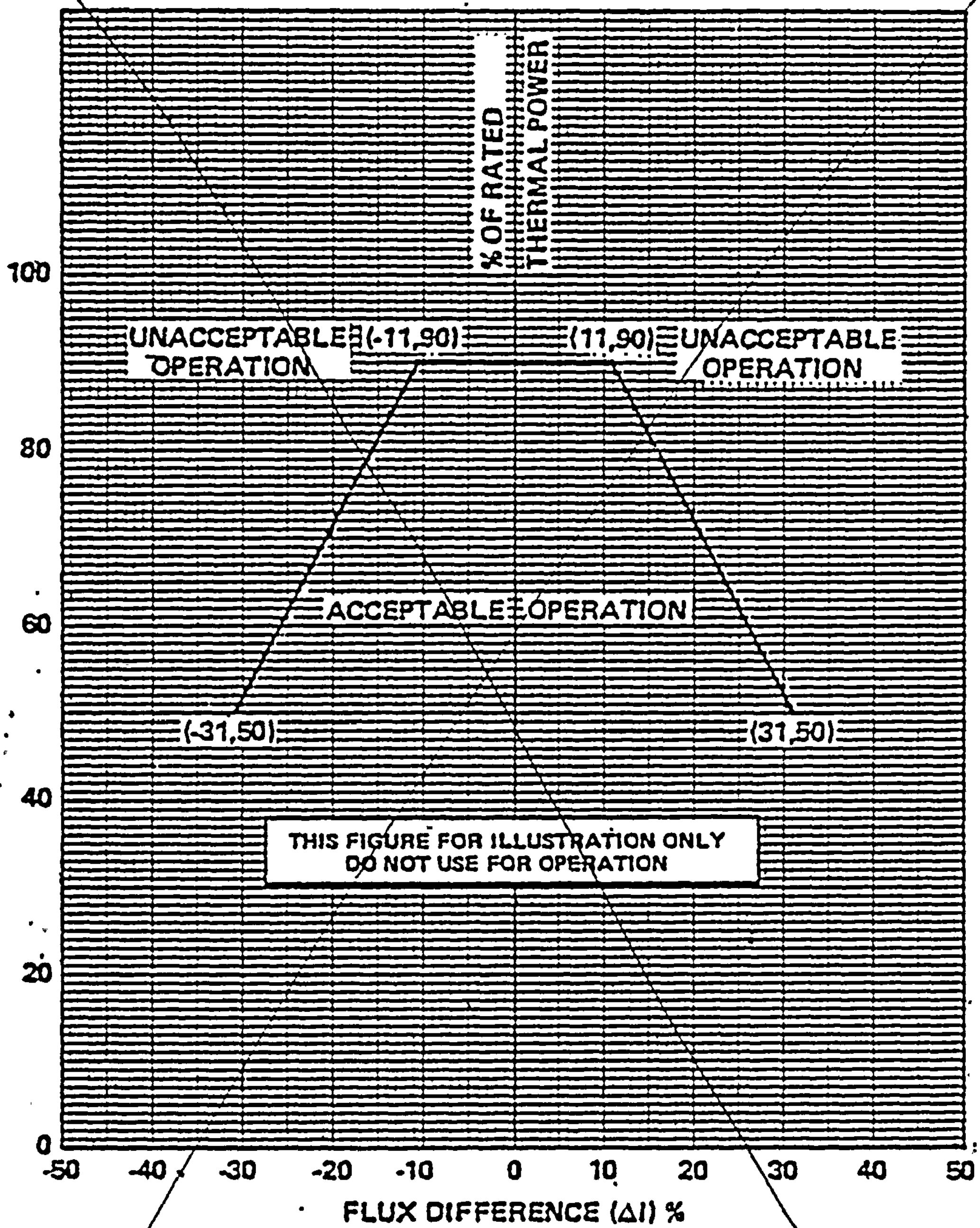

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~~FIGURE 3.2-1~~

~~AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER~~

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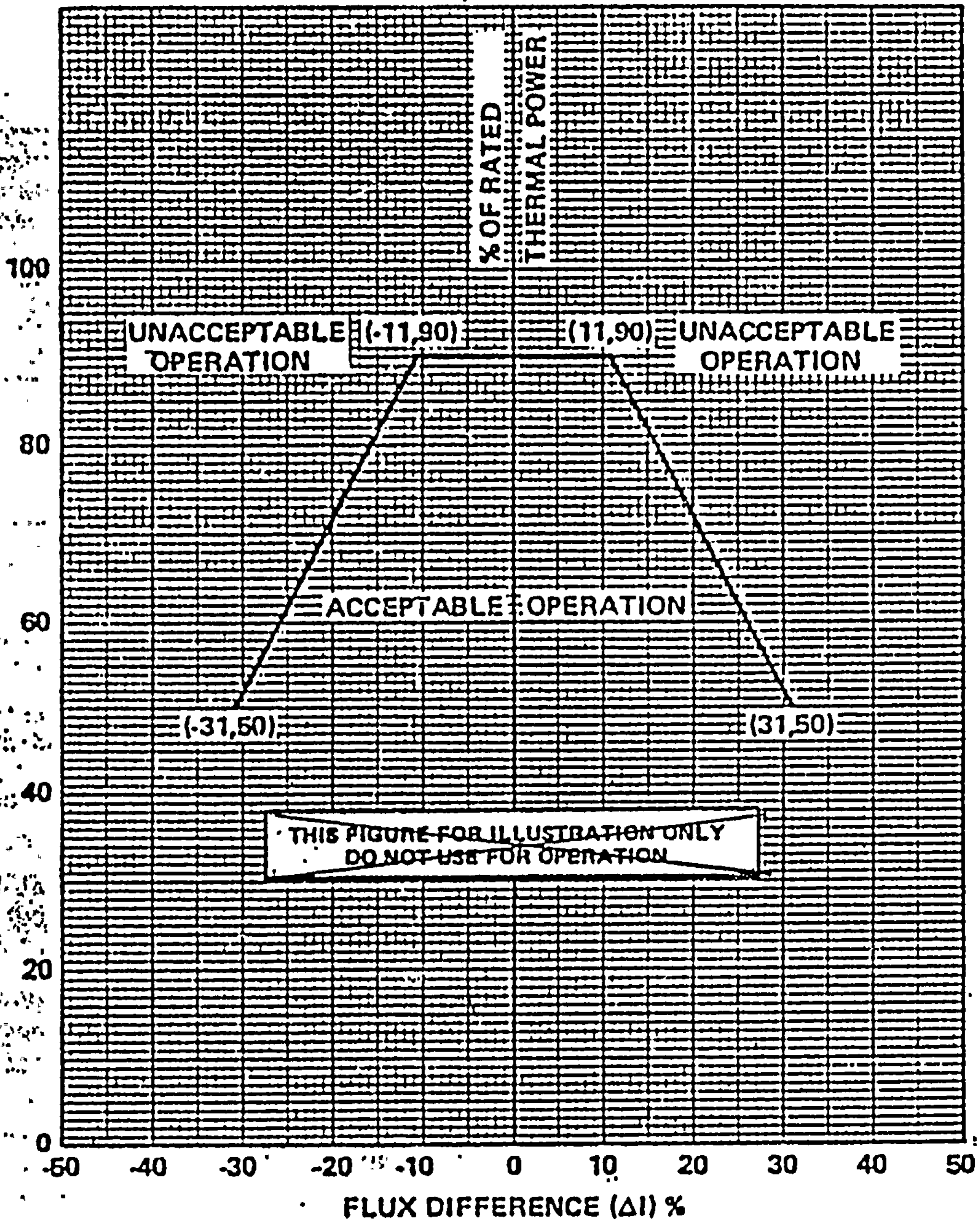

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FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

~~2.7~~ 2.32

$F_Q(Z) \le [\frac{\cancel{2.32}}{P}]\ [K(Z)]$ for $P > 0.5$

~~4.2~~ 4.64

$F_Q(Z) \le [\cancel{(4.64)}]\ [K(Z)]$ for $P \le 0.5$

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the function obtained from Figure [3.2-2] for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

~~a. Comply with either of the following ACTIONS:~~

~~1.~~ Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit, or

~~2. Reduce THERMAL POWER as necessary to meet the limits of Specification [3.2.6] using the Axial Power Distribution Monitoring Systems (APDMS) with the latest incore map and updated R. [APDMS plants only]~~

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

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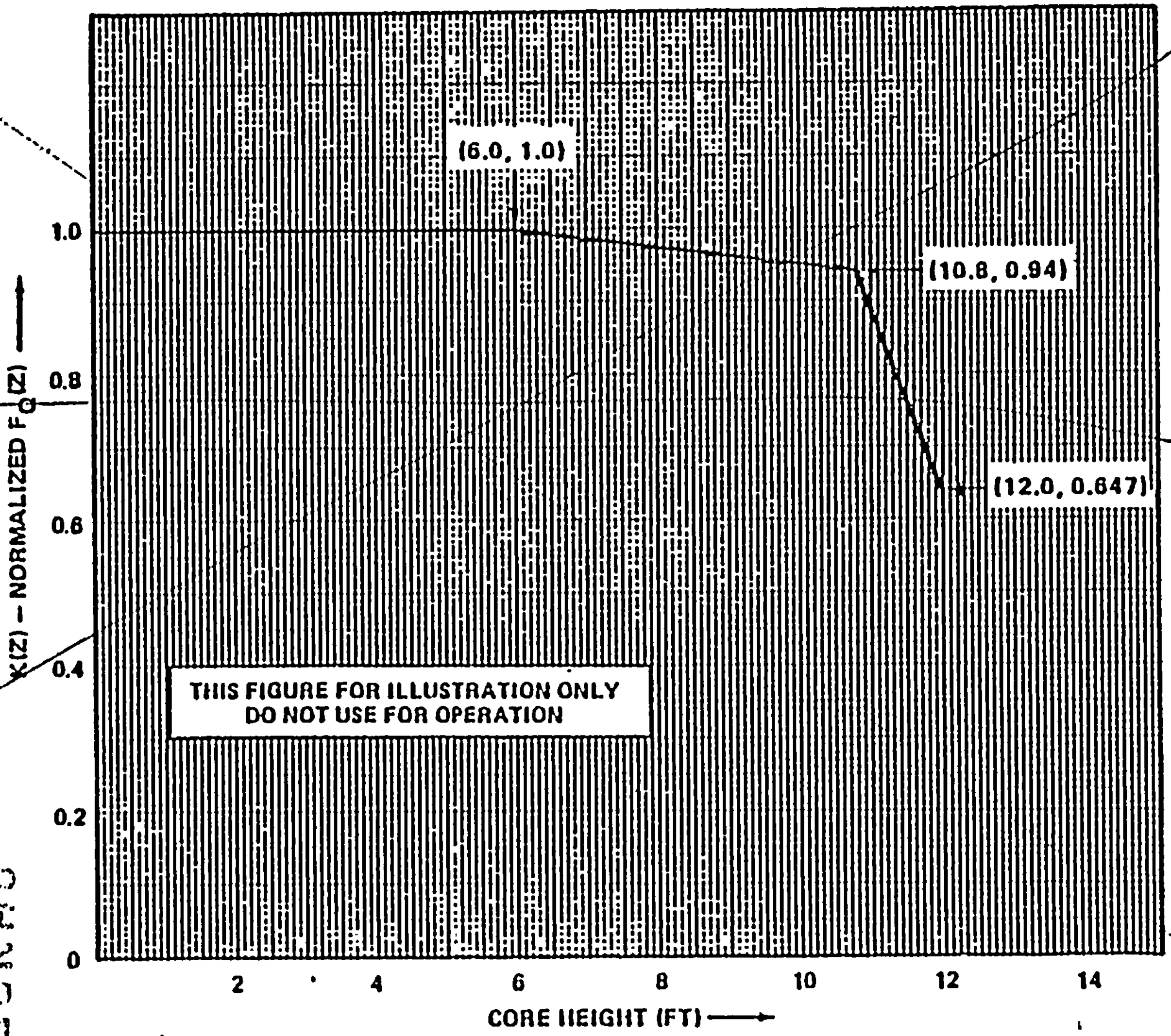

FIGURE 3.2-2

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

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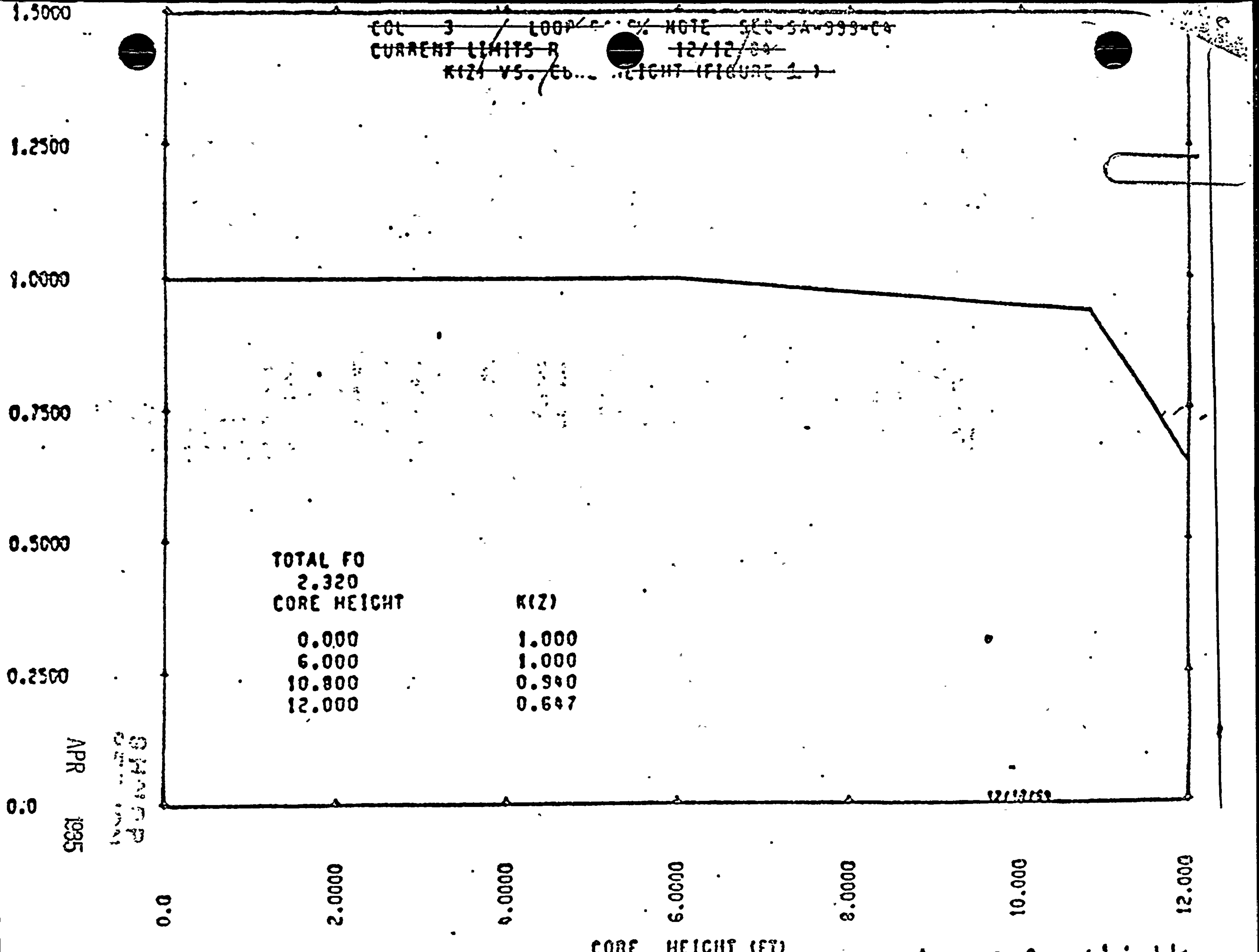

FIGURE 3-2-2. K(z)-Normalized $F_Q(z)$ as a Function of Core Height

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SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER,

~~b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,~~

b ~~c.~~ Comparing the F_{xy} measured ~~computed~~ (F_{xy}^{m}) obtained in Specification 4.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2d ~~e~~. and e ~~f~~., below, and

2) The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP}\ [1+0.2(1-P)],$$

Where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

c ~~d.~~ Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^{M} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^{M} compared to F_{xy}^{RTP} and F_{xy}^{L} either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{M} was last measured ~~determined~~, or

b) At least once per 31 Effective Full Power Days (EFPD), whichever occurs first.

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SURVEILLANCE REQUIREMENTS (Continued)

2) When the F_{xy}^{CM} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{CM} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

d ~~e~~. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.~~6~~9;

e ~~f~~. The F_{xy} limits of Specification 4.2.2.2~~e~~d., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2%, and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

f ~~g~~. With F_{xy}^{CM} exceeding F_{xy}^{L}:

1) The $F_Q(Z)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}. ~~and (for plants with $F_Q(Z)$ less than 2.32 and using APDMS)~~

2) The effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

shall be maintained greater than or equal to 30×10⁴ gpm

3.2.3 The ~~combination of~~ indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained ~~within the region of allowable operation shown on Figure 3.2-3 for four-loop operation.~~ less than or equal to 1.0

three

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since the values listed above ~~Figure 3.2-3~~ includes penalties of 2.4% ~~for undetected feedwater venturi fouling of [0.1]% and~~ for flow measurement uncertainties ~~of [2.1]% for flow~~ and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

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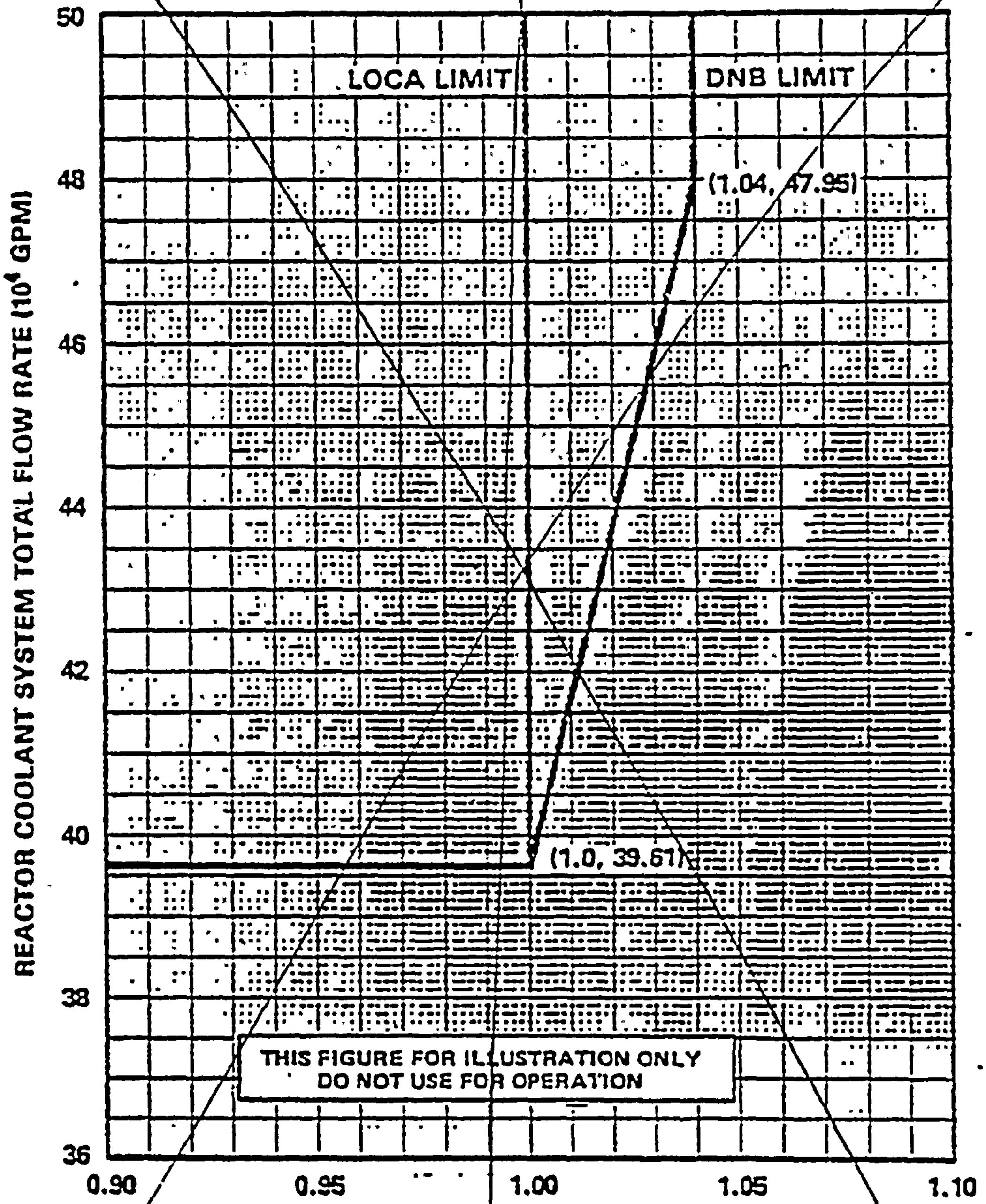
PENALTIES OF 0.1% FOR UNDETECTED FEEDWATER VENTURI FOULING AND MEASUREMENT UNCERTAINTIES OF 2.1% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F^N_{\Delta H}$ ARE INCLUDED IN THIS FIGURE.



$\cancel{R = F^N_{\Delta H}/1.49\ (1 + 0.3(1-P))}$

~~FIGURE 3.2-3~~

~~RCS TOTAL FLOW RATE VERSUS R - FOUR LOOPS IN OPERATION~~

SEE NEXT PAGE

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate determined by process computer readings or digital voltmeter measurement and R shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

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3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

   a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

   b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exceptions Specification 3.10.2.

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

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POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits: ~~shown on Table 3.2-1:~~

a. Indicated Reactor Coolant System T_{avg} ≤ LATER°F, and

b. Indicated Pressurizer Pressure ≥ LATER psig.*

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its indicated limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters ~~of Table 3.2-1~~ SHOWN IN SPECIFICATION 3.2.5 shall be verified to be within its limits at least once per 12 hours.

*THIS LIMIT IS NOT APPLICABLE DURING EITHER A THERMAL POWER RAMP IN EXCESS OF 5% RATED THERMAL POWER PER MINUTE OR A THERMAL POWER STEP INCREASE IN EXCESS OF 10% RATED THERMAL POWER.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS N Loops in Operation	 N-1 Loops in Operation & Loop Stop Valves Open	 N-1 Loops in Operation & Loop Stop Valves Closed
Indicated Reactor Coolant System T_{avg}	$\leq$ [581]°F	$\leq$ [569]°F	$\leq$ [570]°F
Indicated Pressurizer Pressure	$\geq$ [2220] psia*	$\geq$ [2220] psia*	$\geq$ [2220]* psia

*Limit not applicable during either a THERMAL POWER ramp in excess of [5%] of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of [10]% of RATED THERMAL POWER.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

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TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	~~11~~ 10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	~~5~~ 10
c. Shutdown	2	0	1	3, 4, 5	5
7. Overtemperature ΔT					
~~a. Four Loop Plant~~					
~~Four Loop Operation~~	~~4~~	~~2~~	~~3~~	~~1, 2~~	~~6#~~
~~Three Loop Operation~~	~~4~~	~~1**~~	~~3~~	~~1, 2~~	~~9~~
~~b. Three Loop Plant~~					
Three Loop Operation	3	2	2	1, 2	7#
~~Two Loop Operation~~	~~3~~	~~1**~~	~~2~~	~~1, 2~~	~~9~~

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Overpower ΔT					
~~a. Four Loop Plant~~					
~~Four Loop Operation~~	~~4~~	~~2~~	~~3~~	~~1, 2~~	~~6#~~
~~Three Loop Operation~~	~~4~~	~~1**~~	~~3~~	~~1, 2~~	~~9~~
~~b. Three Loop Plant~~					
~~Three Loop Operation~~	3	2	2	1, 2	7#
~~Two Loop Operation~~	~~3~~	~~1**~~	~~2~~	~~1, 2~~	~~9~~
9. Pressurizer Pressure--Low					
~~a. Four Loop Plant~~	~~4~~	~~2~~	~~3~~	~~1~~	~~6#~~
~~b. Three Loop Plant~~	3	2	2	1	7#
10. Pressurizer Pressure--High					
~~a. Four Loop Plant~~	~~4~~	~~2~~	~~3~~	~~1, 2~~	~~6#~~
~~b. Three Loop Plant~~	3	2	2	1, 2	7#
11. Pressurizer Water Level--High	3	2	2	1	7#
12. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop.	1	7#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
13. Steam Generator Water Level--Low-Low	3 ~~4~~/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	7 ~~6~~#
14. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feedwater flow mismatch in each stm. gen.	1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feedwater flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	7#
15. Undervoltage--Reactor Coolant Pumps					
~~a. Four Loop Plant~~	~~4-1/bus~~	~~2~~	~~3~~	~~1~~	~~6#~~
~~b. Three Loop Plant~~	3-1/bus	2	2	1	7#
16. Underfrequency--Reactor Coolant Pumps					
~~a. Four Loop Plant~~	~~4-1/bus~~	~~2~~	~~3~~	~~1~~	~~6#~~
~~b. Three Loop Plant~~	3-1/bus	2	2	1	7#
17. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1	7#
b. Turbine ~~Stop~~ THROTTLE Valve Closure	4	4	1	1	~~12#~~ 6#

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Safety Injection Input from ESF	2	1	2	1, 2	10
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
~~d. Power Range Neutron Flux, P-9~~	~~4~~	~~2~~	~~3~~	~~1~~	~~8~~
d ~~e~~. Power Range Neutron Flux, P-10	4	2	3	1,2	8
e ~~f~~. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
20. Reactor Trip Breakers	2	1	2	1, 2	~~10~~ 9
	2	1	2	3*, 4*, 5*	~~11~~ 10
21. Automatic Trip and Interlock Logic	2	1	2	1, 2	~~10~~ 9
	2	1	2	3*, 4*, 5*	~~11~~ 10

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers ~~happen to be~~ are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

**The channel(s) associated with the trip functions derived from the out-of-service reactor coolant loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to [85]% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, ~~restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, suspend all operations involving positive reactivity changes and verify Valves ________ are closed and secured in position within the next hour.~~

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, ~~STARTUP and/or~~ POWER OPERATION may proceed provided ~~the following conditions are satisfied:~~

~~a.~~ the inoperable channel(s) is placed in the tripped condition within 1 hour~~, and~~.

~~b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.~~

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ~~ANALOG~~ CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

VERIFY COMPLIANCE WITH THE SHUTDOWN MARGIN REQUIREMENTS OF SPECIFICATION 3.1.1.1 or 3.1.1.2, AS APPLICABLE, WITHIN ONE HOUR AND AT LEAST ONCE PER 12 HOURS THEREAFTER.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

~~ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.~~

ACTION ~~10~~ 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION ~~11~~ 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

~~ACTION 12 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.~~

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TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	≤ 4 seconds*
8. Overpower ΔT	≤ 4 seconds*
9. Pressurizer Pressure--Low	≤ 2 seconds
10. Pressurizer Pressure--High	≤ 2 seconds
11. Pressurizer Water Level--High	N.A.

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel. ~~(This provision is not applicable to CPs docketed after January 1, 1978. See Regulatory Guide 1.118, November 1977.)~~

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TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of ~~Reactor Coolant~~ Flow ~~--Low~~	
a. Single Loop (Above P-8)	≤ [1] second
b. Two Loops (Above P-7 and below P-8)	≤ [1] second
13. Steam Generator Water Level--Low-Low	≤ [2] seconds
14. Steam Generator Water Level-Low Coincident with Steam/Feedwater Flow Mismatch	N.A.
15. Undervoltage - Reactor Coolant Pumps	≤ [1.5] seconds
16. Underfrequency - Reactor Coolant Pumps	≤ [0.6] second
17. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
18. Safety Injection Input from ESF	N.A.
19. Reactor Trip System Interlocks	N.A.
20. Reactor Trip Breakers	N.A.
21. Automatic Trip and Interlock Logic	N.A.

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TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	~~M~~ Q(13)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	~~M~~ Q(13)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	~~M~~ Q(13)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	~~M~~ Q(13)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1), ~~M~~ Q(13)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S	R(4, 5, ~~12~~)	S/U(1), ~~M(9)~~ Q(9, 13)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	S	R(13)	~~M~~ Q(13)	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	~~M~~ Q(13)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	M	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	M	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	M	N.A.	N.A.	1
12. Loss of ~~Reactor Coolant~~ Flow ~~--Low~~	S	R	~~M~~ Q(13)	N.A.	N.A.	1

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	1, 2
14. Steam Generator Water Level -- Low Coincident with Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	~~M~~ Q(10,13)	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	~~M~~ Q(10,13)	N.A.	1
17. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	~~M~~ Q(13)	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	~~M(8)~~ Q(8,13)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	~~M(8)~~ Q(8,13)	N.A.	N.A.	1
~~d. Power Range Neutron Flux, P-9~~	~~N.A.~~	~~R(4)~~	~~M(8)~~	~~N.A.~~	~~N.A.~~	~~1~~

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19 ~~20~~. Reactor Trip System Interlocks (Continued)						
d ~~f~~. Power Range Neutron Flux, P-10	N.A.	R(4)	Q(8,13) ~~M(8)~~	N.A.	N.A.	1, 2
e ~~g~~. Turbine Impulse Chamber Pressure, P-13	N.A.	R	Q(8,13) ~~M(8)~~	N.A.	N.A.	1
20 ~~21~~. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 10, 11)	N.A.	1, 2, 3*, 4*, 5*
21 ~~22~~. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* Only if the Reactor Trip System breakers are ~~happen to be~~ closed and the Control Rod Drive System is capable of rod withdrawal.

** Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

*** Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) With power greater than or equal to the Interlock Setpoint the required ~~ANALOG~~ CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) Quarterly ~~Monthly~~ surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. ~~Monthly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to (an increase of twice the count rate within a 10-minute period).~~

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include ~~independent verification of the Undervoltage and Shunt trips~~.

~~(12) At least once per 18 months during shutdown, verify that on a simulated Boron Dilution Doubling test signal the normal CVCS discharge valves will close and the centrifugal charging pumps suction valves from the RWST will open within [30] seconds.~~

(1~~3~~2) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

SEPARATE TESTS WHICH VERIFY THAT THE UNDERVOLTAGE AND SHUNT TRIPS ACTUATE THE REACTOR TRIP BREAKERS.

(13) STAGGERED TEST BASIS is required if surveillance interval greater than one month is used.

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3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $Z + R + S \leq TA$

   Where:

   Z = The value from Column Z of Table 3.3-4 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

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SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

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TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment VENTILATION ISOLATION, PHASE A CONTAINMENT ISOLATION, ~~AND~~ START AUXILARY FEEDWATER SYSTEM MOTOR DRIVEN PUMP, ~~ACTUATION~~) ~~Cooling Fans, and Essential Service Water)~~ START CONTAINMENT FAN COOLERS, START EMERGENCY SERVICE WATER PUMPS, AND START EMERGENCY SERVICE WATER BOOSTER PUMPS)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	~~15~~ 18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, ~~4~~	14
c. Containment Pressure--High-1	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure--Low	3 ~~4~~	2	~~3~~ 2	1, 2, 3#	~~20*~~ 15*
e. ~~Differential~~ STEAM LINE ~~Pressure Between~~ PRESSURE ~~Steam Lines--High~~ LOW				1, 2, 3~~##~~#	
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~3/steam line~~	~~2/steam line any steam line~~	~~2/steam line~~		~~15*~~
~~Three Loops Operating~~	3/~~operating~~ steam line	2 ~~1****~~/steam line ~~any operating steam line~~ ANY ~~AND~~ 1/3 STEAM LINES	2/~~operating~~ steam line		~~16~~ 15*

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water) (Continued)					
2) Three Loop Plant					
Three Loops Operating	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line		15*
Two Loops Operating	3/operating steam line	2***/steam line twice in either operating steam line	2/operating steam line		16
f. Steam Line Pressure--Low				1, 2, 3**	
1) Four Loop Plant					
Four Loops Operating	1 pressure loop	1 pressure any 2 loops	1 pressure any 3 loops		15*
Three Loops Operating	1 pressure/operating loop	1*** pressure in any operating loop	1 pressure in in any 2 operating loops		16

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water) (Continued)					
2) Three Loop Plant					
Three Loops Operating	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 2 loops		15*
Two Loops Operating	1 pressure/ loop	1*** pressure in any operating loop	1 pressure any operating loop		16
2. Containment Spray					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	~~15~~ 18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-3	4	2	3	1, 2, 3	~~17~~ 16

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18 ~~19, 24~~
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation	SEE 2a ABOVE FOR MANUAL CONTAINMENT SPRAY INITIATING FUNCTIONS AND REQUIREMENTS.				
1) Manual CONTAINMENT SPRAY Initiation	~~2~~	~~1 with 2 coincident switches~~	~~2~~	~~1, 2, 3, 4~~	~~19~~
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-3	~~4~~	~~2~~	~~3~~	~~1, 2, 3~~	~~17~~
	SEE 2c ABOVE FOR CONTAINMENT PRESSURE HIGH-THREE INITIATING FUNCTIONS AND REQUIREMENTS				
c. CONTAINMENT VENTILATION ~~Purge and Exhaust~~ Isolation					
1) Manual CONTAINMENT SPRAY ~~Initiation~~	SEE 2a ABOVE FOR ALL MANUAL CONTAINMENT SPRAY INITIATING FUNCTIONS AND REQUIREMENTS				
	~~2~~	~~1~~	~~2~~	~~1, 2, 3, 4,~~	~~18~~
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4, 6**	18, 25
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
4) CONTAINMENT RADIOACTIVITY-HIGH	SEE TABLE 3.3-6 ITEM 1				
5) MANUAL PHASE "A" ISOLATION	SEE 3a1 ABOVE FOR ALL MANUAL PHASE A ISOLATION INITIATING FUNCTIONS AND REQUIREMENTS.				

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN Steam Line Isolation					
a. Manual Initiation					
1) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	2~~4~~3
2) System	2	1	2	1, 2, 3	2~~3~~2
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	2~~2~~1
c. Containment Pressure--High-2	3	2	2	1, 2, 3	15*
d. ~~Steam Flow in Two Steam Lines--High~~ STEAM LINE PRESSURE LOW	SEE 1e ABOVE FOR ALL STEAM LINE PRESSURE LOW INITIATING FUNCTIONS AND REQUIREMENTS			1, 2, 3	
~~1) Four Loop Plant~~					
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		15*
Three Loops Operating	2/operating steam line	1***/any operating steam line	1/operating steam line		16
2) Three Loop Plant					
Three Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		15*
Two Loops Operating	2/operating steam line	1***/any operating steam line	1/operating steam line		16
e. NEGATIVE STEAM LINE PRESSURE RATE-HIGH	3/STEAM LINE	2 IN ANY STEAM LINE	2/STEAM LINE	3**, 4**	15*

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
Steam Flow in Two Steam Lines--High					
Coincident With:					
T_{avg}--Low-Low				1, 2, 3	
1) Four Loop Plant					
Four Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 3 loops		15*
Three Loops Operating	1 T_{avg}/operating loop	1*** T_{avg} in any operating loop	1 T_{avg} in any two operating loops		16
2) Three Loop Plant					
Three Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops		15*
Two Loops Operating	1 T_{avg}/operating loop	1*** T_{avg} in any operating loop	1 T_{avg} in any operating loop		16

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
e. Steam Line Pressure--Low				1, 2, 3**	
1) Four Loop Plant					
Four Loops Operating	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 3 loops,		15*
Three Loops Operating	1 pressure/ operating loop	1*** pressure in any operating loop	1 pressure in any 2 operating loops		16
2) Three Loop Plant					
Three Loops Operating	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 2 loops		15*
Two Loops Operating	1 pressure/ operating loop	1*** pressure in any operating loop	1 pressure any operating loop		16
f. Steam Line Pressure - Negative Rate--High				3****	
1) Four Loop Plant					
Four Loops Operating	3/steam line	2/steam line any steam line	2/steam line		15*
Three Loops Operating	3/operating steam line	2/steam line in any operating steam line	2/steam line in each operating steam line		16

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
~~4. Steam Line Isolation (Continued)~~					
~~2) Three Loop Plant~~					
~~Three Loops Operating~~	~~3/steam line~~	~~2/steam line any steam line~~	~~2/steam line~~		~~15~~
~~Two Loops Operating~~	~~3/operating steam line~~	~~2/steam line in any operating steam line~~	~~2/steam line in each operating steam line~~		~~16~~
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	~~25~~ 24
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2	~~20*~~ 19*
c. SAFETY INJECTION	SEE I ABOVE FOR ALL SAFETY INJECTION INITIATING FUNCTIONS AND REQUIROMENTS.				
6. Auxiliary Feedwater					
~~a. Manual Initiation~~	~~2~~	~~1~~	~~2~~	~~1, 2, 3~~	~~23~~
a ~~b.~~ Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	~~22~~ 21
b ~~c.~~ Stm. Gen. Water Level--Low-Low					
1) Start Motor-Driven Pumps	3 ~~4~~/stm. gen.	2/stm. gen. in any ~~operating~~ stm. gen.	2 ~~3~~/stm. gen. in each ~~operating~~ stm. gen.	1, 2, 3	~~20*~~ 15*

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
2) Start Turbine-Driven Pump	3 ~~2~~/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	2 ~~3~~/stm. gen. in each operating stm. gen.	1, 2, 3	15* ~~20*~~
~~d. Undervoltage RCP Start Turbine-Driven Pump~~	~~4-1/bus~~	~~2~~	~~3~~	~~1, 2~~	~~20*~~
c ~~e~~. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
d ~~f~~. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	SEE 9 BELOW FOR ALL LOSS OF OFFSITE POWER INITIATING FUNCTIONS AND REQUIREMENTS				
	~~2~~	~~1~~	~~2~~	~~1, 2, 3~~	~~19~~
e ~~g~~. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps ~~and Turbine-Driven Pump~~	1 ~~2~~/pump	1/pump	1/pump	1, 2	~~19~~ 18
f ~~h. Suction Transfer on Low Pressure~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3~~	~~19~~
7. SAFETY INJECTION ~~Automatic~~ Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
i. STEAM LINE DIFFERENTIAL PRESSURE-HIGH	3/STEAM LINE	1/STEAM LINE COMPARED TO 1/STEAM LINE ON 2 OTHER STEAM LINES	2/STEAM LINE	1, 2, 3	15*
ii. MAIN STEAM ISOLATION	SEE 4 ABOVE FOR ALL STEAM LINE ISOLATION INITIATING FUNCTIONS AND REQUIREMENTS				

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. ~~Automatic~~ Safety Injection Switchover to Containment Sump (Continued)					
b. RWST Level--Low-Low	4	2	3	1, 2, 3, 4	~~17~~ 16
~~Coincident With:~~ ~~Containment Sump Level--High~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3, 4~~	~~17~~
~~and~~ COINCIDENT WITH Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
9 ~~8~~. Loss of Offsite Power					
a. ~~4 kV Bus Under~~ 6.9KV Emergency Bus-Primary ~~voltage Loss of Voltage~~	3 ~~4~~/bus	2/bus	2 ~~3~~/bus	1, 2, 3, 4	15 ~~20~~*
b. ~~4 kV Bus Undervoltage~~ 6.9KV Emergency Bus-Secondary ~~Grid Degraded Voltage~~	3 ~~4~~/bus	2/bus	2 ~~3~~/bus	1, 2, 3, 4	15 ~~20~~*
10. (SEE NEXT PAGE)					
11 ~~10~~ ~~9~~. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
~~b. Low Low T_{avg}, P-12~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3~~	~~21~~
b ~~c~~. Reactor Trip, P-4	2	2	2	1, 2, 3	23
c ~~d~~. Steam Generator Water Level, P-14	~~3/stm. gen.~~	~~2/stm. gen. in any operating stm. gen.~~	~~2/stm. gen. in each operating stm. gen.~~	~~1, 2, 3~~	~~21~~
	SEE ITEM 5.B ABOVE FOR ALL P-14 INITIATING FUNCTIONS AND REQUIREMENTS.				
8. CONTAINMENT SPRAY SWITCHOVER TO CONTAINMENT SUMP					
a. AUTOMATIC ACTUATION LOGIC AND ACTUATION RELAYS	2	1	2	1,2,3,4	14
b. RWST--LOW-LOW	See ITEM 7.b above for all RWST--LOW-LOW INITIATING FUNCTIONS AND REQUIREMENTS.				
COINCIDENT WITH CONTAINMENT SPRAY	SEE. ITEM 2 ABOVE FOR ALL CONTAINMENT SPRAY INITIATING FUNCTIONS AND REQUIREMENTS.				

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TABLE 3.3-3 (CONTINUED)

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. CONTROL ROOM ISOLATION AND CONTROL ROOM EMERGENCY FILTRATION ACTUATION					
a. AUTOMATIC ACTUATION LOGIC AND ACTUATION RELAYS	2	1	2	All	14
b. Safety INJECTION	See 1 ABOVE FOR SAFETY INJECTION INITIATING FUNCTIONS AND REQUIREMENTS				
c. HIGH RADIATION	SEE TABLE 3.3-6, ITEM 3				
D. HIGH CHLORINE	SEE SPECIFICATION 3.3.3.7				

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TABLE 3.3-3 (Continued)

TABLE NOTATIONS

*The provisions of Specification 3.0.4 are not applicable.

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

** DURING CORE ALTERATIONS OR MOVEMENT OF IRRADIATED FUEL ~~**Trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) Setpoint.~~ WITHIN THE CONTAINMENT. REFER TO SPECIFICATION 3.9.9.

***The channel(s) associated with the protective functions derived from the out of service reactor coolant loop shall be placed in the tripped mode.

****Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ~~ANALOG~~ CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

~~ACTION 16 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.~~

ACTION ~~17~~ 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION ~~18~~ 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the Containment Purge ~~supply~~ ~~EXHA~~ MAKEUP and exhaust valves are maintained closed, WHILE IN MODES 1,2,3 & 4 (REFER TO SPECIFICATION 3.6.1.7). FOR MODE 6 REFER TO SPECIFICATION 3.9.4.

TABLE 3.3-3 (Continued)

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ACTION STATEMENTS (Continued)

ACTION ~~19~~ 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION ~~20~~ 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION ~~21~~ 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION ~~22~~ 21 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION ~~23~~ 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION ~~24~~ 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification ~~[~~3.7.1.5~~]~~.

ACTION ~~25~~ 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION-25 DURING CORE ALTERATIONS OR MOVEMENT OF ~~SPENT~~ IRRADIATED FUEL WITHIN THE CONTAINMENT, COMPLY WITH THE ACTION STATEMENT ~~OR~~ OF SPECIFICATION 3.9.9.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection ~~(Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water)~~					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic AND ACTUATION RELAYS	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High 1	2.7 ~~[3.0]~~	~~[0.71]~~	~~[1.5]~~	≤ 3.0 ~~[3.6]~~ psig	≤ 3.6 ~~[3.86]~~ psig
d. Pressurizer Pressure--Low	18.8 ~~[13.1]~~	14.4 ~~[10.71]~~	~~[1.5]~~	≤ ~~[1850]~~ psig	≤ 1836 ~~[1839]~~ psig
~~e. Differential Pressure Between Steam Lines--High~~	~~[3.0]~~	~~[0.87]~~	~~[1.5/ 1.5]~~	~~≤ [97] psi~~	~~≤ [106] psi~~
e ~~f.~~ Steam Line Pressure--Low	16.6 ~~[20.0]~~	14.8 ~~[10.71]~~	~~[1.5]~~	≤ 601 ~~[675]~~ psig	≤ 590.4 ~~[635]~~ psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3	3.6 ~~[3.0]~~	~~[0.71]~~	~~[1.5]~~	≤ 10.0 ~~[12.05]~~ psig	≤ 11.0 ~~[12.31]~~ psig

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Phase "B" Isolation					
1) Manual CONTAINMENT SPRAY Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure--High-3	SEE ITEM 2c ABOVE FOR CONTAINMENT PRESSURE HIGH-3 TRIP SETPOINTS AND ALLOWABLE VALUES. ~~[3.0]~~	~~[0.71]~~	~~[1.5]~~	~~≤ [12.05] psig~~	~~≤ [12.31] psig~~
c. CONTAINMENT VENTILATION ~~Purge and Exhaust~~ Isolation					
1) Manual CONTAINMENT SPRAY ~~Initiation~~	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
5) MANUAL PHASE "A" ISOLATION	NA	N.A.	NA.	N.A.	N.A.
4) CONTAINMENT RADIOACTIVITY-High	SEE TABLE 3.3-6 ITEM 1				

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. MAIN ~~M~~ Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2	2.7 ~~[3.0]~~	~~[0.71]~~	~~[1.5]~~	≤3.0 ~~[≤6.35]~~ psig	≤3.6 ~~[≤6.61]~~ psig
~~d. Steam Flow in Two Steam Lines--High, Coincident with~~	~~[20.0]~~	~~[13.16]~~	~~[1.5/ 1.5]~~	~~≤ A function defined as follows: A ΔP corresponding to 40% of full steam flow between 0% and 20% load and then a ΔP increasing linearly to 110% of full load.~~	~~≤ A function defined as follows: A ΔP corresponding to 44% of full steam flow between 0% and load and then a ΔP increasing linearly to a ΔP corresponding to 114.0% of full steam flow at full load.~~
~~T_{avg}--Low-Low~~	~~[4.0]~~	~~[1.12]~~	~~[1.2]~~	~~≤[553]°F~~	~~≤[550.6]°F~~
~~d e. Steam Line Pressure--Low~~	~~16.6 [20.0]~~	~~14.8 [10.71]~~	~~[1.5]~~	~~601 ≤[675] psig~~	~~590.4 ≤[635] psig*~~
e ~~f~~. Steam Line Pressure - Negative Rate--High	~~[8.0]~~	~~[0.5]~~	[0]	-100 ≤~~[110]~~ psi/s	-111.5 ≤ ~~[121.6]~~ psi/s**

SEE ITEM 1.e ABOVE FOR STEAMLINE PRESSURE -- LOW TRIP SETPOINTS AND ALLOWABLE VALUES.

~~#8?~~ SHEARON HARRIS UNIT 1

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--High-High (P-14)	7.6 ~~[5.0]~~	4.28 ~~[2.18]~~	~~[1.5]~~	≤~~[~~82.4~~]~~% of narrow range instrument span.	≤~~[~~84.2~~]~~% of narrow range instrument span.
c. SAFETY INJECTION	SEE 1 ABOVE FOR SI SETPOINTS AND ALLOWABLE VALUES.				
6. Auxiliary Feedwater					
~~a. Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~
a ~~b~~. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b ~~c~~. Steam Generator Water Level--Low-Low	19.2 ~~[30.0]~~	18.2 ~~[27.18]~~	~~[1.5]~~	≥ 38.3 ~~[32.2]~~% of narrow range instrument span.	≥ 37.8 ~~[30.4]~~% of narrow range instrument span.
~~d.~~ ~~Loss of Offsite Power~~ ~~Undervoltage - RCP~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≤ [70]% RCP bus voltage.~~	~~≥ [69]% RCP bus voltage.~~

~~SEE ITEM 9 BELOW~~

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~~#575~~ SHEARON HARRIS UNIT 1

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
c ~~e~~. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
d ~~f~~. Loss-of-Offsite Power	SEE ITEM 9 BELOW FOR ALL LOSS OF OFFSITE POWER TRIP SETPOINTS AND ALLOWABLE VALUES. ~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≥ [4800]V~~	~~≥ [4692]V~~
e ~~g~~. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
f. (see next page)					
~~h. Suction Transfer on Low Pressure~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≤ [442] ft~~	~~≤ [441] ft~~
7. ~~Automatic~~ SAFETY INJECTION Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low Coincident With	N.A.	N.A.	N.A.	≤ ~~[18]~~ 38.5%	≤ ~~[15]~~ 37.4%
~~Containment Sump Level~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≤ [30] in. above [680] ft~~	~~≤ [32.5] in. above [680] ft~~
~~and~~ 8. (SEE NEXT PAGE) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values				
9 ~~8~~. Loss of Offsite Power					
a. ~~4 kV Bus Undervoltage (Loss-of-Voltage)~~ 6.9kV EMERGENCY BUS UNDERVOLTAGE - PRIMARY	N.A.	N.A.	N.A.	≤ ~~[5760]~~ ~~4761~~ LATER volts with a ≤ ~~[0.25]~~ ~~.5~~ LATER second time delay.	≤ ~~[5652]~~ ~~4692~~ LATER volts with a ≤ ~~[0.275]~~ ~~.575~~ LATER second time delay.
b. ~~4 kV Bus Undervoltage (Grid Degraded Voltage)~~ 6.9 KV EMERGENCY BUS UNDERVOLTAGE - SECONDARY	N.A.	N.A.	N.A.	≤ ~~[6576]~~ ~~5885~~ LATER volts with a ≤ ~~[3.3]~~ 60 second time delay (without Safety Injection)	≤ ~~[6511]~~ ~~5816~~ LATER volts with a ≤ ~~[3.3]~~ 61.8 second time delay (without Safety Injection)

SEE NEXT PAGE FOR CONTINUATION OF 9.

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TABLE 3.3-4 INSERT (Page 3/4 3-34)

6.f STEAMLINE DIFFERENTIAL PRESSURE - HIGH COINCIDENT WITH	5.0	2.07	3.0	≤ 100 psi	≤ 119.6 psi
MAIN STEAMLINE ISOLATION	SEE 4 ABOVE FOR ALL MAINSTEAMLINE ISOLATION TRIP SETPOINTS AND ALLOWABLE VALUES				
8. CONTAINMENT SPRAY SWITCH-OVER TO CONTAINMENT SUMP					
a. AUTOMATIC ACTUATION LOGIC AND ACTUATION RELAYS	NA	NA	NA	NA	NA
b. RWST - LOW - LOW	SEE 7.A ABOVE FOR ALL RWST - LOW - LOW SETPOIN AND ALLOWABLE VALUES				
COINCIDENT WITH CONTAINMENT SPRAY	SEE 2 ABOVE FOR ALL CONTAINMENT SPRAY SETPOINTS AND ALLOWABLE VALUES.				
9. LOSS OF OFFSITE POWER (CONTINUED)				≤ ~~5805~~ LATER volts with a ~~15~~ LATER second time delay with Safety Injection	≤ ~~5816~~ LATER volts with a ~~15.45~~ LATER second time delay with Safety Injectio

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ ~~[1985]~~ 2000 psig	≤ ~~[1996]~~ 2014 psig
~~not P-11~~	~~N.A~~	~~N.A.~~	~~N.A.~~	~~* ≥ 2000 psig~~	~~≥ 1986 psig~~
~~b. Low-Low T_{avg}, P-12~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≥ [553]°F~~	~~≥ [550.6]°F &~~
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Water Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

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TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq [50]$ seconds and $\tau_2 \geq [5]$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is less than or equal to ~~[50]~~ seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

50

~~(LATER)~~

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1.	Manual Initiation	
	a. Safety Injection (ECCS)	N.A.
	b. Containment Spray	N.A.
	c. Phase "A" Isolation	N.A.
	~~d. Phase "B" Isolation~~	~~N.A.~~
	d ~~e.~~ CONTAINMENT VENTILATION ~~Purge and Exhaust~~ Isolation	N.A.
	e ~~f.~~ Steam Line Isolation	N.A.
	~~g. Feedwater Isolation~~	~~N.A.~~
	~~h. Auxiliary Feedwater~~	~~N.A.~~
	~~i. Essential Service Water~~	~~N.A.~~
	~~j. Containment Cooling Fans~~	~~N.A.~~
	~~k. Control Room Isolation~~	~~N.A.~~
	f. ~~l.~~ Reactor Trip	N.A.
	g. ~~m.~~ Start Diesel Generator	N.A.
2.	Containment Pressure--High-1	
	a. Safety Injection (ECCS)	≤ ~~{27}~~ 27(1)/{12}(~~3~~ 4)
	1) Reactor Trip	≤ {2}
	2) Feedwater Isolation	≤ {7}~~(3)~~
	3) CONTAINMENT Phase "A" Isolation	≤ {17}(2)/{27}(1)
	4) CONTAINMENT VENTILATION ~~Purge and Exhaust~~ Isolation	≤ ~~{25}(1)/{10}(2)~~ 5(5)
	5) Auxiliary Feedwater MOTOR DRIVEN PUMPS	≤ {60}
	6) EMERGENCY ~~Essential~~ Service Water	≤ ~~{32}(1)/{47}(2)~~ (LATER)
	7) Containment FAN COOLERS ~~Cooling Fans~~	≤ ~~{55}(1)/{40}(2)~~ (LATER)
	8) Control Room Isolation	N.A.
	9) Start Diesel Generator	≤ ~~{10}~~ (LATER)

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	≤ ~~[27]~~ 27 (1)/~~[12]~~ 4 (~~5~~)
1) Reactor Trip	≤ ~~[~~2~~]~~
2) Feedwater Isolation	≤ ~~[~~7~~]~~(3)
3) CONTAINMENT ~~Phase "A"~~ Isolation	≤ ~~[~~17~~]~~(2)/~~[~~27~~]~~(1)
4) CONTAINMENT VENTILATION ~~Purge and Exhaust~~ Isolation	≤ ~~[25](1)/[10](2)~~ 5 (5)
5) Auxiliary Feedwater MOTOR DRIVEN PUMPS	≤ ~~[~~60~~]~~
6) EMERGENCY ~~Essential~~ Service Water PUMPS	≤ ~~[47](1)/[32](2)~~ (LATER)
7) Containment FAN COOLERS ~~Cooling Fans~~	≤ ~~[55](1)/[40](2)~~ (LATER)
8) Control Room Isolation	N.A.
~~9) Start Diesel Generators~~	~~≤ [10] (LATER)~~
~~4. Differential Pressure Between Steam Lines--High~~	
~~a. Safety Injection (ECCS)~~	~~≤ [22](4)/[12](5)~~
~~1) Reactor Trip~~	~~≤ [2]~~
~~2) Feedwater Isolation~~	~~≤ [7](3)~~
~~3) Phase "A" Isolation~~	~~≤ [17](2)/[27](1)~~
~~4) Purge and Exhaust Isolation~~	~~≤ [25](1)/[10](2)~~
~~5) Auxiliary Feedwater~~	~~≤ [60]~~
~~6) Essential Service Water~~	~~≤ [32](2)/[47](1)~~
~~7) Containment Cooling Fans~~	~~≤ [55](1)/[40](2)~~
~~8) Control Room Isolation~~	~~N.A.~~
~~9) Start Diesel Generators~~	~~≤ [10]~~
4 ~~5.~~ MAIN Steam Line Pressure--Low	
a. Safety Injection (ECCS)	≤ ~~[~~12~~]~~ 4 (~~5~~)/~~[~~22~~]~~ 3 (~~4~~)
1) Reactor Trip	≤ ~~[~~2~~]~~
2) Feedwater Isolation	≤ ~~[~~7~~]~~(3)
3) CONTAINMENT ~~Phase "A"~~ Isolation	≤ ~~[~~17~~]~~(2)/~~[~~27~~]~~(1)
4) CONTAINMENT VENTILATION ~~Purge and Exhaust~~ Isolation	≤ ~~[25](1)/[10](2)~~ 5 (5)
5) Auxiliary Feedwater MOTOR DRIVEN PUMPS	≤ ~~[~~60~~]~~
6) EMERGENCY ~~Essential~~ Service Water PUMPS	≤ ~~[32](2)/[47](1)~~ (LATER)
7) Containment Cooling Fans	≤ ~~[55](1)/[40](2)~~ (LATER)

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4 ~~8~~. Steam Line Pressure--Low (Continued)	
8) Control Room Isolation	N.A.
9) Start Diesel Generators	≤ ~~[10]~~ (LATER)
b. Steam Line Isolation	≤ ~~[9]~~(3) 7
5 ~~6~~. Containment Pressure--High-3	
a. Containment Spray	≤ ~~[45](2)/[57](1)~~ (LATER)
b. Phase "B" Isolation	≤ ~~[65](1)/[75](2)~~ (LATER)
6 ~~7~~. Containment Pressure--High-2	
Steam Line Isolation	≤ ~~[9](3)~~ 7
~~8. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--LowLow~~	
~~Steam Line Isolation~~	~~≤ [9]~~(3)
7 ~~8~~. Steam Line Pressure - Negative Rate--High	
Steam Line Isolation	≤ ~~[9](3)~~ 7
8 ~~10~~. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ ~~[~~2.5~~]~~
b. Feedwater Isolation	≤ ~~[~~7~~](3)~~
9 ~~11~~. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ ~~[~~60~~]~~
b. Turbine-Driven Auxiliary Feedwater Pump	≤ ~~[~~60~~]~~
~~10.~~ ~~12. Undervoltage RCP~~	
~~Turbine-Driven Auxiliary Feedwater Pump~~	~~≤ [60]~~
13. Loss-of-Offsite Power	
Turbine-Driven Auxiliary Feedwater Pump	≤ ~~[60]~~ (LATER)
10 ~~14~~. Trip of All Main Feedwater Pumps	
All Auxiliary Feedwater Pumps	N.A.
11. High DIFFERENTIAL STEAM GENERATOR PRESSURE COINCIDENT WITH MAIN STEAMLINE ISOLATION SIGNAL	(Later)
a. Isolate Auxiliary Feedwater to the Affected Steam Generator	

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
~~15. Suction Transfer on Low Pressure~~	
~~Auxiliary Feedwater (Suction Supply Automatic Realignment)~~	~~≤ [13]~~
12 ~~16.~~ RWST Level--Low-Low	
a. Automatic Switchover to Containment Sump	~~N.A.~~
~~b.~~ Coincident with ~~Containment Sump Level--High and~~ Safety Injection ~~(Automatic Switchover to Containment Sump)~~	(LATER) ~~≤[250](2)/[265](1)~~
b. Automatic Switchover to Containment Sump Coincident wiht Containment Spray.	(LATER)
13 ~~17. Loss of Power~~ Offsite	
~~14 a. 6.9 kV Emergency Bus Undervoltage - Primary (Loss of Voltage)~~	~~≤ [10] (LATER)~~
~~b. 6.9 kV Emergency Bus Undervoltage (Grid - Secondary Degraded Voltage)~~	~~≤ [10] (LATER)~~
14. Containment High Radiation	
a. Containment Isolation	LATER

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TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

~~(3) Air-operated valves.~~

3 ~~(4)~~ Diesel generator starting and sequence loading delay included. RHR pumps not included.

4 ~~(5)~~ Diesel generator starting and sequence loading delays not included. RHR pumps not included.

(5) Isolation of Normal Containment Purge; THIS VALUE IS NOT APPLICABLE TO Pre-Entry Containment Purge, WHICH IS ASSUMED TO BE OPERATING ONLY IN MODES 5 OR 6.

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TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection ~~(Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water)~~ Use titles from Table 3.3-3								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, ~~4~~
c. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~e. Differential Pressure Between Steam Lines-High~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
e. ~~f.~~ Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
C. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Containment Spray Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Ventilation ~~Purge and Exhaust~~ Isolation								
1) Manual Containment Spray Initiation	SEE ITEM 2.a. ABOVE FOR ALL MANUAL CONTAINMENT SPRAY SURVEILLANCE REQUIREMENTS ~~N.A.~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4~~
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1),(2)	M(1),(2)	Q (2)	1, 2, 3, 4, 6#
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) Containment Radioactivity-High	See Table 4.3-3 Item 1 For Surveillance Requirements							
5) Manual Phase A	See 3.a.1 Above for all Manual Phase A Isolation Surveillance Requirements							

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SHEARON HARRIS UNIT 1

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~d. Steam Flow in Two Steam Lines-High Coincident With~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A~~	~~N.A.~~	~~N.A~~	~~1, 2, 3~~
~~T_{avg}-Low-Low~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A~~	~~1, 2, 3~~
d ~~e~~. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e ~~f~~. Steam Line Pressure-Negative Rate-High	S	R	M	N.A.	N.A.	N.A.	N.A.	3, 4
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. SAFETY INJECTION	SEE ITEM 1 ABOVE FOR ALL SAFETY INJECTION SURVEILLANCE REQUIREMENTS							
6. Auxiliary Feedwater								
~~a. Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
a ~~b~~. Automatic Actuation and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3

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~~W-STS~~ Shearon Harris Unit 1

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)								
b ~~c~~. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
~~d. Undervoltage - RCP~~	~~N.A.~~	~~R.~~	~~N.A.~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2~~
c ~~e~~. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
d ~~f~~. Loss-of-Offsite Power	See Item 8 below for Loss of Offsite Surveillance Requirements ~~N.A.~~	~~R~~	~~N.A.~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A~~	~~1, 2, 3~~
e ~~g~~. Trip of All Main Feed-water Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
f. Isolation i. High Differential Steam Generator Pressure Coincident With ii & Main Steam Isolation Signal ~~h. Suction Transfer on Low Pressure~~	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
	See 4 above for Main Steam Isolation Surveillance Requirements.							
7. Safety Injection ~~Automatic~~ Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low Coincident With	S	R	M	N.A.	N.A.	N.A.	N.A	1, 2, 3, 4
~~Containment Sump Level-High~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4~~
~~and~~								
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

8. (See Next page for INSERT)

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INSERT FOR PAGE 3/4 3-45

8. CONTAINMENT SPRAY SWITCHOVER TO CONTAINMENT SUMP

a. RWST Low Low Level	See 7.a above for RWST Low Low Level Surveillance Requirements
Coincident with b. Containment Spray	See 2 above for Containment Spray Surveillance Requirements

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INSERT FOR TABLE 4.3-2

... Page 3/4 3-46

10.

CONTROL ROOM ISOLATION
AND CONTROL ROOM EMERGENCY
FILTRATION ACTUATION

a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	All

b. Safety Injection — See 1 above for all Safety Injection Surveillance Requirements

c. High Radiation — See Table 4.3-3 Item 3

d. High Chlorine — See Specification 4.3.3.7

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SHEARON HARRIS - UNIT 1 ~~STS~~

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9 ~~8~~. Loss of OFFSITE Power								
a. 6.9kV EMERGENCY ~~4 kV~~ Bus Undervoltage ~~(Loss of Voltage)~~ PRIMARY	N.A.	R	N.A	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 6.9kV EMERGENCY ~~4 kV~~ Bus Undervoltage ~~(Grid Degraded Voltage)~~ SECONDARY	N.A.	R.	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
10. (Refer to INSERT ON NEXT PAGE)								
11 ~~9~~. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~b. Low-Low T_{avg}, P-12~~	~~N.A.~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
b ~~c~~. Reactor Trip, P-4	N.A.	N.A	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
c ~~d~~. Steam Generator Water Level, P-14	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~M(1)~~	~~M(1)~~	~~Q~~	~~1, 2, 3~~

See Item 5.b above for P-14 Surveillance Requirements

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) The surveillance requirements of Specification 4.9.9 apply during CORE ALTERATIONS or movement of irradiated fuel within the Containment

~~##~~ # During CORE ALTERATIONS or movement of irradiated fuel within the containment.

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3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ~~ANALOG~~ CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

REFER TO NEXT PAGES FOR DATA.

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High	1	2	All	≤ [2] mR/h	26
b. RCS Leakage Detection					
1) Particulate Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
2) Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
2. Purge and Exhaust Ventilation					
a. Particulate Radioactivity	1	2	All	*	26
b. Gaseous Radioactivity	1	2	All	*	26
3. Fuel Storage Pool Areas					
a. Radioactivity-High					
Gaseous Radioactivity	1	2	**	≤ [2] mR/h	27
b. Criticality-Radiation Level	1	2	***	≤ 15 mR/h	28
4. Control Room					
a. Air Intake-Radiation Level	1/intake	2/intake	All	≤ [2] mR/h	27
b. Control Room Atmosphere Radiation-High	1	2	All	≤ [2] mR/h	27

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TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

INSTRUMENT	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP	ACTION
1. Containment Atmosphere-					
a. Containment Ventilation Isolation Signal Area Monitors	2	3	1,2,3,4,6	(later)	27
b. Gaseous Radioactivity RCS Leakage Detection-	N.A.	1	1,2,3,4	N.A.	26 ~~25~~
c. Particulate Radioactivity RCS Leakage Detection	N.A.	1	1,2,3,4	N.A.	26 ~~25~~
2. Spent Fuel Pool Area-Fuel Handling Building Emergency Exhaust Actuation					
a. Fuel Handling Building Operating Floor - South Network	2***	2	**	(later)	28 ~~26~~
b. Fuel Handling Building Operating Floor - North Network	2***	3	*	(later)	28 ~~26~~
3. Control Room Outside Air Intakes-					
a. Normal Outside Air Intake Isolation	1	2	All	(later)	29 ~~28~~

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TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

INSTRUMENT	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP	ACTION
b. Emergency Outside Air Intake Isolation-South Intake	1	2	All	(later)	~~28~~ 29
c. Emergency Outside Air Intake Isolation-North Intake	1	2	All	(later)	~~28~~ 29

*With irradiated fuel in the Northend Spent Fuel Pool or transfer of irradiated fuel from or to a spent fuel shipping cask.

**With irradiated fuel in the Southend Spent Fuel Pool or New Fuel Pool.

***Each channel consists of 3 detectors with 2 of 3 logic, a channel is OPERABLE when 2 of the detectors cause a channel to trip by being in any of the following modes: exceeded high radiation set point, failed detector, test mode.

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TABLE 3.3-6 (Continued)

TABLE NOTATIONS

~~* Must satisfy Specification 3.11.2.1 requirements.~~

~~** With irradiated fuel in the fuel storage pool areas.~~

~~*** With fuel in the fuel storage pool areas.~~

ACTION STATEMENTS

ACTION ~~26~~ 7 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the all containment purge ~~and~~ makeup and exhaust isolation valves are maintained closed or, as applicable; comply with the ACTION requirements of Specification 3.9.9.

ACTION ~~27~~ 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour ~~isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.~~ initiate isolation of the respective air intake. With no outside air intakes available maintain operation of the Control Room Emergency Filtration System in the recirculation mode of operation.

~~ACTION 28 - With less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool areas.~~

ACTION ~~29~~ 6 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 28 WITH THE number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

REFER TO NEXT 2 PAGES FOR DATA

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere Radioactivity-High	S	R	M	All
b. RCS Leakage Detection				
1) Particulate Radioactivity	S	R	M	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	M	1, 2, 3, 4
2. Purge and Exhaust Ventilation				
a. Particulate Radioactivity	S	R	M	All
b. Gaseous Radioactivity	S	R	M	All
3. Fuel Storage Pool Areas				
a. Radioactivity-High-Gaseous Radioactivity	S	R	M	**
b. Criticality-Radiation Level	S	R	M	*
4. Control Room				
a. Air Intake Radiation Level	S	R	M	All
b. Control Room Atmosphere Radiation-High	S	R	M	All

TABLE NOTATIONS

* With fuel in the fuel storage pool area.
** With irradiated fuel in the fuel storage pool areas.

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~~INSERT C - INSTRUMENTATION~~

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment Atmosphere-				
a. Containment Ventilation Isolation Signal	S	R	M	1,2,3,4,6
b. Gaseous Radioactivity-RCS Leakage Detection	S	R	M	1,2,3 & 4
c. Particulate Radioactivity-RCS Leakage Detection	S	R	M	1,2,3 & 4
2. Spent Fuel Pool Area-Fuel Handling Building Emergency Exhaust Actuation Signal				
a. Fuel Handling Building Operating Floor-South Network	S	R	M	**
b. Fuel Handling Building Operating Floor-North Network	S	R	M	*

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~~INSERT 8 — INSTRUMENTATION~~

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Control Room Normal Outside Air Intakes				
a. Normal Outside Air Intake Isolation	S	R	M	All
b. Emergency Outside Air Intake Isolation-South Intake	S	R	M	All
c. Emergency Outside Air Intake Isolation-North Intake	S	R	M	All

*With irradiated fuel in the Northend Spent Fuel Pool or transfer of irradiated fuel from or to a spent fuel shipping cask.

**With irradiated fuel in the Southend Spent Fuel Pool or New Fuel Pool.

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MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 38 ~~75% of~~ the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$, and F_{xy}.

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INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7~~*~~ shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and ~~ANALOG~~ CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to ~~[~~0.01~~]~~ g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features ~~important to safety.~~ DESIGNED TO SEISMIC CATEGORY I REQUIREMENTS

~~[*The instrumentation may be shared with additional units at a common site provided seismic instrumentation and corresponding Technical Specifications meet the recommendations of Regulatory Guide 1.12, Revision 1, April 1974.]~~

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TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. Containment Mat (EL. 221 Ft)	.01 – 1.0g	1**
b. Containment (EL. 286 ft)	.01 – 1.0g	1**
c. Diesel Fuel Oil Storage	.01 – 1.0g	1**
~~d.~~ Tank Building (EL 242 ft)	~~—~~	~~1~~
2. Triaxial Peak Accelerograph~~s~~ Recorders		
a. Reactor Coolant Pipe (Loop 1)	± 10g	1
b. Steam Generator 1A Pedestal (EL 238 ft)	± 2g	1
c. Reactor Auxilary Building (EL 236 ft)	± 10g	1
~~d.~~		~~1~~
~~e.~~		~~1~~
3. Triaxial Seismic Switches		
a. Starter Unit for TimeHistory	.01g ** (H or V)	1*
~~b.~~ Accelerograph System-Containment		~~1*~~
~~c.~~ Mat (EL 221 ft)		~~1*~~
b ~~d.~~ Triaxial Seismic Switch-Containment Mat (EL 221 ft)	(V) .113g **, (E-W) .119g **, (N-S) .092g **	1*
4. Triaxial Response-Spectrum Recorders		
a. Steam Generator 1B Pedestal (EL 238 ft)	± 2g	1*
b. Reactor Auxilary Building (EL 216 ft)	± 2g	1
c. Diesel Fuel Oil Storage Tank Bldg. (EL	± 2g	1
d. Containment Building (EL 221 ft)	± 2g	1*
~~e.~~		~~1~~
~~f.~~		~~1~~

*With ~~reactor~~ Main control room indication

** Setpoints for seismic switches, with direction designations H- Horizontal, V- Vertical, E- East, W- West, N- North, S- South.

*** With Main Control Room Recording..

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TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST
1. Triaxial Time-History Accelerographs			
a. Containment Mat (El 221 ft)	M*	R	SA ***
b. Containment (EL 286 ft)	M*	R	SA ***
c. ~~Diesel Fuel Oil Storage~~	~~M*~~	~~R~~	~~SA~~
~~d.~~ Tank Building (EL. 242 ft)	M*	R	SA ***
2. Triaxial Peak Accelerographs			
a. Reactor Coolant Pipe (Loop 1)	N.A.	R	N.A.
b. Steam Generator 1A Pedestal (EL 238 ft)	N.A.	R	N.A.
c. Reactor Auxilary Building (EL 236 ft)	N.A.	R	N.A.
~~d.~~	~~N.A.~~	~~R~~	~~N.A.~~
~~e.~~	~~N.A.~~	~~R~~	~~N.A.~~
3. Triaxial Seismic Switches			
a. Starter Unit for Time History ~~**~~	~~M~~	~~R~~	~~SA~~
~~b.~~ Accelerograph Syst-Containment Mat ** (EL 221 ft)	M	R	SA ***
b. ~~c.~~ Triaxial Seismic Switch Containment ~~**~~	~~M~~	~~R~~	~~SA~~
~~d.~~ Mat (EL 221 ft) **	M	R	SA ***
4. Triaxial Response-Spectrum Recorders			
a. Steam Generator 1B Pedestal ~~**~~	~~M~~ N.A.	R	~~SA~~ N.A.
b. Reactor Auxilary Building (Passive) (EL 216 ft)	N.A.	R	~~SA~~ N.A.
c. Diesel Fuel Oil Storage Tank Building (Passive) (EL 242 ft)	N.A.	R	~~SA~~ N.A.
d. Containment Building (Active) (EL 221 ft) **	~~N.A.~~ M	R	SA ***
~~e.~~	~~N.A.~~	~~R~~	~~SA~~
~~f.~~	~~N.A.~~	~~R~~	~~SA~~

*Except seismic ~~trigger~~ starter unit.

**With Main ~~reactor~~ Control Room ~~indications~~ Alarms

*** THE BISTABLE TRIP SETPOINT NEED NOT BE DETERMINED DURING THE PERFORMANCE OF A CHANNEL OPERATIONAL TEST

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METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-5.

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TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. Wind Speed		
~~a.~~	Nominal Elev. 12.5 METERS	1
~~b.~~	Nominal Elev. 61.4 METERS	1
2. Wind Direction		
~~a.~~	Nominal Elev. 12.5 METERS	1
~~b.~~	Nominal Elev. 61.4 METERS	1
3. Air Temperature ~~ΔT~~ - DIFFERENTIAL TEMPERATURE		
~~a.~~	~~Nominal Elev.~~ 11.0 METERS TO 59.9 METERS	1
~~b.~~	~~Nominal Elev.~~	1

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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 12.5 M	D	SA
b. Nominal Elev. 61.4 M	D	SA
2. Wind Direction		
a. Nominal Elev. 12.5 M	D	SA
b. Nominal Elev. 61.4 M	D	SA
3. Air Temperature ~~- ΔT~~ Differential Temperature		
~~a. Nominal Elev.~~ 11.0 M to 59.9 M	D	SA
~~b. Nominal Elev.~~	D	SA

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INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5.a The Remote Shutdown System ~~transfer switches, power, controls and~~ monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With one or more Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b inoperable, restore the inoperable switch(s)/circuit(s) to OPERABLE status within ~~7~~ 30 days, or be in HOT STANDBY within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit including the actuated components required by 3.3.3.5.b shall be demonstrated OPERABLE at least once per 18 months.

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3.3.3.5.b The Safe Shutdown Division 2 Transfer Switches, Auxiliary Control Panel Controls, Auxiliary Transfer Panel Controls and local controls for the OPERABILITY of those components required to (1) remove decay heat via auxiliary feedwater flow and Steam Generator power operated relief valve flow, (2) Control RCS inventory and (3) control RCS pressure shall be OPERABLE.

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REFER TO NEXT PAGE

TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM

INSTRUMENT [Illustrational only]	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Power Range Neutron Flux		2	2
2. Intermediate Range Neutron Flux		2	2
3. Source Range Neutron Flux		2	2
4. Reactor Trip Breaker Indication		1/trip breaker	1/trip breaker
5. Reactor Coolant Temperature - Average		2	2
6. Reactor Coolant Flow Rate		2	2
7. Pressurizer Pressure		2	2
8. Pressurizer Level		2	2
9. Steam Generator Pressure		2/stm. gen.	2/stm. gen.
10. Steam Generator Water Level		2/stm. gen.	2/stm. gen.
11. Control Rod Position Limit Switches		1/insertion limit switch/ rod	1/insertion limit switch/rod
12. RHR Flow Rate		2	2
13. RHR Temperature		2	2
14. Auxiliary Feedwater Flow Rate		2	2

TRANSFER SWITCHES [Illustrational Only]	SWITCH LOCATION
1. Auxiliary Feedwater Control	
2. Safe Shutdown Equipment Power	
a. Auxiliary Feedwater	
b. Charging	
c. Pressurizer Heaters	
d. Valves	
3. CVCS Makeup Flow Control	
4. Diesel Generator Control	
5. Electrical Distribution System Control	

CONTROL CIRCUITS [Illustrational Only]	SWITCH LOCATION
1. Auxiliary Feedwater Flow	
2. Pressurizer Heaters	
3. CVCS Makeup Flow	
4. Diesel Generator	
5. Electrical Distribution System	

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TABLE 3.3-9A

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	TOTAL NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	Reactor Trip Switchgear	1/Trip Breaker	1/Trip Breaker
2. RCS Hot Leg Temperature	ACP*	2	1
3. RCS Cold Leg Temperature	ACP*	2	1
4. Pressurizer Pressure	ACP*	2	1
5. Pressurizer Level	ACP*	2	1
6. Steam Generator Pressure	ACP*	1/Steam Generator	1/Steam Generator
7. Steam Generator Level - Wide Range	ACP*	1/Steam Generator	1/Steam Generator
8. RHR Flow Rate	ACP*	1/RHR Train	1/RHR Train
9. Auxiliary Feedwater Flow Rate	ACP*	1/Steam Generator	1/Steam Generator
10. Condensate Storage Tank Level	ACP*	2	1
~~INSERT X~~			
~~*Auxiliary Control Panel~~			
11. REACTOR COOLANT SYSTEM PRESSURE - WIDE RANGE	ACP*	2	1
12 ~~Source Range~~ Flux Monitor	(LATER)	1	1
13. CHARGING HEADER FLOW	ACP*	1	1
14. a AFW Turbine Inlet - Pump Discharge ΔP or		1	1
b. Auxiliary Feedwater Turbine Speed	ACP*		
15. BORIC ACID TANK LEVEL	ACP*	1	1

* ACP = Auxiliary Control Panel

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TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
~~1. Power Range Neutron Flux~~	~~M~~	~~Q~~
~~2. Intermediate Range Neutron Flux~~	~~M~~	~~N.A.~~
12 ~~13~~. Source Range Neutron Flux	M	N.A.
1. ~~4~~. Reactor Trip Breaker Indication	M	N.A.
2. ~~5~~. Reactor Coolant System Hot Leg Temperature ~~Coolant Temperature - Average~~	M	R
3. Reactor Coolant System Cold Leg Temperature	M	R
~~6. Reactor Coolant Flow Rate~~	~~M~~	~~R~~
4 ~~7~~. Pressurizer Pressure	M	R
5 ~~8~~. Pressurizer Level	M	R
6 ~~9~~. Steam Generator Pressure	M	R
7 ~~10~~. Steam Generator Water Level - Wide Range	M	R
~~11. Control Rod Position Limit Switches~~	~~M~~	~~R~~
8 ~~12~~. RHR Flow Rate	M	R
15. Boric Acid Tank Level	M	R
~~13. RHR Temperature~~	~~M~~	~~R~~
9 ~~14~~. Auxiliary Feedwater Flow Rate	M	R
10. Condensate Storage Tank Level	M	R
11. Reactor Coolant System Pressure - Wide Range	M	R
13. Charging Header Flow	M	R
14. a. Auxiliary FW Turbine Inlet-Pump Discharge ΔP	M	R
b. Auxiliary Feedwater Turbine Speed	M	R

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ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

Containment Post-LOCA Monitors

Turbine Building Vent Stack Monitor, ... d Waste Processing Building E...aust System (Vent 1 & Vent 5) Monitors

b. With the number of OPERABLE accident monitoring instrumentation channels except the ~~unit vent high range noble gas~~ Plant Vent Stack Monitor - High Range Monitor, the Main Steam Line ~~relief high range radiation~~ monitors, ~~the containment atmosphere high range radiation monitor, and the reactor coolant radiation level monitor~~ less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

c. With the number of OPERABLE channels for the ~~unit vent high range noble gas~~ Plant Vent Stack Monitor - High Range Monitor, or the Main steam Line ~~relief high range radiation~~ monitors ~~or the containment atmosphere high range radiation monitor, or the reactor coolant radiation level monitor~~ less than required by the Minimum Channels OPERABLE requirements, initiate an alternate method of monitoring the appropriate parameter(s), within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT ~~[Illustrational Only]~~	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure a. Narrow Range	2	1
b. Wide Range	2	1
2. Reactor Coolant ~~Outlet~~ Hot Leg Temperature ~~- T_{HOT} (Wide Range)~~	2	1
3. Reactor Coolant ~~Inlet~~ Cold Leg Temperature ~~- T_{COLD} (Wide Range)~~	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
~~10. Boric Acid Tank Solution Level~~	~~2~~	~~1~~
~~11~~ 10. Auxiliary Feedwater Flow Rate	~~2~~ 1/steam generator	1/steam generator
~~12~~ 11. Reactor Coolant System Subcooling Margin Monitor	2	1
~~13~~ 12. PORV Position Indicator*	2/valve	1/valve
~~14~~ 13. PORV Block Valve Position Indicator**	2/valve	1/valve
~~15~~ 14. Safety Valve Position Indicator	2/valve	1/valve
~~16~~ 15. Containment Water Level ~~(Narrow Range)~~	2	1
~~17~~ 16. Containment Water Level ~~(Wide Range)~~	2	1

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TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT [Illustrational Only]	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
7 ~~18~~. In Core Thermocouples	4/core quadrant	2/core quadrant
8 ~~19~~. Plant ~~Unit~~ Vent Stack Monitor - High Range Noble Gas Monitor	N.A.	1
22 ~~20~~ ~~20~~. Main Steam Line ~~Steam Relief - High Range Radiation~~ Monitors	N.A.	1/steam line
19. ~~21~~. Containment POST - LOCA ~~Atmosphere - High Range Radiation~~ Monitors	N.A.	1
20 ~~21~~ ~~22~~. Reactor Vessel ~~Water~~ Level Indication System	2	1
~~23. Reactor Coolant Radiation Level Monitor~~	~~N.A.~~	~~1~~
21 ~~22~~. Containment Spray NaOH Tank Level	2	1
23 ~~24~~. Turbine Building Vent Stack Monitor	N.A.	1
24 ~~25~~. Waste Processing Building Exhausts System Vent 5	N.A	1
Vent 5A	N.A.	1
25 ~~26~~. Condensate Storage Tank Level	2	1

* Not applicable if the associated block value is in the closed position.

** Not applicable if the valve is verified in the closed position and power is removed.

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT ~~[Illustrational Only]~~	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure a. Narrow Range	M	R
b. Wide Range	M	R
2. Reactor Coolant Hot Leg ~~Outlet~~ Temperature - ~~T_{HOT}~~ ~~(Wide Range)~~	M	R
3. Reactor Coolant Cold Leg ~~Inlet~~ Temperature - ~~T_{COLD}~~ ~~(Wide Range)~~	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
~~10. Boric Acid Tank Solution Level~~	~~M~~	~~R~~
~~11~~ 10. Auxiliary Feedwater Flow Rate	M	R
~~12~~ 11. Reactor Coolant System Subcooling Margin Monitor	M	R
~~13~~ 12. PORV Position Indicator	M	R
~~14~~ 13. PORV Block Valve Position Indicator	M	R
~~15~~ 14. Safety Valve Position Indicator	M	R
~~16~~ 15. Containment Water Level ~~(Narrow Range)~~	M	R
~~17~~ 16. Containment Water Level ~~(Wide Range)~~	M	R

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~~W-STS~~ SHEARON HARRIS - UNIT 2

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT ~~[Illustrational Only]~~	CHANNEL CHECK	CHANNEL CALIBRATION
~~18~~ 17. In Core Thermocouples	M	R
~~19~~ 18. ~~Unit~~ Plant Vent Stack Monitor - High Range Noble Gas Monitor	M	R
22 ~~23~~ ~~20~~. Main Steam Line ~~Steam Relief - High Range Radiation~~ Monitor	M	R
19 ~~21~~. Containment Post-LOCA ~~Atmosphere - High Range Radiation~~ Monitors	M	R*
20 ~~21~~ ~~22~~. Reactor Vessel ~~Water~~ Level Indication System	M	R
~~23. Reactor Coolant Radiation Level Monitor.~~	M	R

*CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

21 ~~22~~. Containment Spray NaOH Tank Level	M	R
~~23~~ ~~24~~ 23 Turbine Building Vent Stack Monitor	M	R
~~25~~ 24 Waste Processing Building Exhaust System Vent 5	M	R
Vent 5A	M	R
~~26~~ 25 Condensate Storage Tank Level	M	R

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INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

LATER

3.3.3.7 Two independent Chlorine Detect~~ion Systems~~ or Trains, with their ~~Alarm/~~Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to ~~5~~ ppm, shall be OPERABLE. EACH TRAIN SHALL CONSIST OF A DETECTOR AT EACH CONTROL ROOM AREA VENTILATION SYSTEM INTAKE (NORMAL AND EMERGENCY) OR A DETECTOR AT THE CHLORINE STORAGE AREA.

APPLICABILITY: All MODES.

ACTION:

a. With one Chlorine Detect~~ion System~~ or or Train inoperable, restore the inoperable system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room ~~Emergency~~ AREA Ventilation System in the recirculation mode of operation.

b. With both Chlorine Detect~~ion Systems~~ or Trains inoperable, within 1 hour initiate and maintain operation of the Control Room ~~Emergency~~ AREA Ventilation System in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each Chlorine Detect~~ion System~~ or shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, an ~~ANALOG~~ CHANNEL OPERATIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

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FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION: REFER TO INSERT ON THE FOLLOWING PAGE

a. ~~With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6).~~ 5

b. ~~With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6).~~

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.8.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

~~4.3.3.8.3 The nonsupervised circuits, associated with detector alarms, between the instrument and the control room shall be demonstrated OPERABLE at least once per 31 days.~~

INSERT FOR PAGE 3/4 3-67

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or ~~(~~monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.~~6)~~5.

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days, or in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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REFER TO FOLLOWING PAGES FOR DATA.

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION [Illustrative]	HEAT (x/y)	TOTAL NUMBER OF INSTRUMENTS* FLAME (x/y)	SMOKE (x/y)
1. Containment **			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
2. Control Room			
3. Cable Spreading			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
4. Computer Room			
5. Switchgear Room			
6. Remote Shutdown Panels			
7. Station Battery Rooms			
8. Turbine			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
9. Diesel Generator			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
10. Safety-Related Pumps			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
11. Fuel Storage			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			

[List all detectors in areas required to ensure the OPERABILITY of safety-related equipment.]

---

*(x/y): x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of Fire Suppression Systems and early warning and notification) instruments.

**The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

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INSERT H - INSTRUMENTATION

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

Zone	Instrument Location	Elevation FT.	Minimum Number of Instruments Operable: Heat (A/B)*	Flame (A/B)*	Smoke (A/B)*
1.0	Containment Building**				
1-C-1-RCP-1A	Reactor Coolant Pump 1A	256.33	9/0	-	-
1-C-1-RCP-1B	Reactor Coolant Pump 1B	256.33	9/0	-	-
1-C-1-RCP-1C	Reactor Coolant Pump 1C	256.33	9/0	-	-
1-C-1-CH~~E~~FA	Airborne Radioactivity Removal Unit 1A	221.0	0/3	-	-
1-C-1-CH~~E~~FB	Airborne Radioactivity Removal Unit 1B	221.0	0/3	-	-
1-C-1-EPA	Electrical Penetration Area 1A	261.0	0/9	-	9/0
1-C-1-EPB	Electrical Penetration Area 1B	261.0	0/9	-	9/0
2.0	Reactor Auxiliary Building				
1-A-1-PA	RHR Pump Room 1A	190.0	0/8	-	-
1-A-1-PB	RHR Pump Room 1B	190.0	0/8	-	-
1-A-2-MP	Misc. Pumps & Equipment	216.0	-	-	24/0
1-A-3-PB	Auxiliary Feed Water Pumps, Component Cooling Water, Pumps & Heat Exchangers	236.0	0/36	-	38/0
1-A-3-COMB	Decontamination Area & Corridor Cable Trays	236.0	0/6	-	10/0
1-A-3-COM~~B~~E	Let Down Heat Exchanger & Corridor Cable Trays	236.0	0/4	-	13/0

*(A/B) A = The number of early warning fire detectors
B = The number of detectors used for actuation of fire suppression system's

** The fire detection instruments located within the Containment Building are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

Zone	Instrument Location	Elevation FT	Minimum Number of Instruments Operable Heat (A/B)	Flame (A/B)	Smoke (A/B)
2.0	Reactor Auxiliary Building (Cont'd)				
1-A-3-COM1	Recycle Holdup Tank Area & Corridor Cable Trays	236.0	0/6	-	15/0
1-A-4-CHLR	HVAC Chiller Equipment Area and Cable Trays	261.0	0/31	-	31/0
1-A-4-COMB	Boric Acid Equipment Area & Corridor Cable Trays	261.0	0/9	-	9/0
1-A-4-COME	Corridor Cable Trays	261.0	0/4	-	9/0
1-A-4-COM1	Corridor Cable Trays	261.0	0/2	-	4/0
1-A-4-CHFA	Charcoal Filter Room 1A	261.0	0/3	-	7/0
1-A-4-CHFB	Charcoal Filter Room 1B	261.0	0/2	-	6/0
1-A-EPA	Electrical Penetration Area SA	261.0	0/10	-	10/0
1-A-EPB	Electrical Penetration Area SB	261.0	0/10	-	10/0
1-A-5-HVA	HVAC Room 1A	286.0	-	-	10/0
1-A-5-HVB	HVAC Room 1B	286.0	-	-	10/0
1-A-SWGRA	Switch Gear Room A	286.0	-	-	13/0
1-A-SWGRB	Switch Gear Room B	286.0	-	-	12/0
1-A-BATA	Battery Room 1A	286.0	-	-	1/0
1-A-BATB	Battery Room 1B	286.0	-	-	1/0
1-A-CSRA	Cable Spreading Room A	286.0	0/20	-	20/0
1-B-CSRB	Cable Spreading Room B	286.0	0/10	-	10/0
1-A-ACP	Auxiliary Control Panel	286.0	-	-	1/0
12-A-6-RT1	Terminal Cabinet Room	305.0	-	-	10/0
12-A-6-RCC1	Rod Control Cabinets Room	305.0	-	-	4/0

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INSERT H - INSTRUMENTATION (Cont.d)

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

Zone	Instrument Location	Elevation FT	Minimum Number of Instruments Operable Heat (A/B)	Flame (A/B)	Smoke (A/B)
3.0	Fuel Handling Building (Cont'd)				
12-A-6-CR1	Control Room	305.0	-	-	13/0
12-A-6-APR 1	Auxiliary Relay Panels	305.0	-	-	4/0
12-A-6-PICR1	Process Instruments & Control Racks	305.0	-	-	5/0
12-A-6-HV7	HVAC Equipment Room	305.0	0/2	-	9/0
3.0	Fuel Handling Building				
5-F-2-FPC	Fuel Pool Cooling Pumps and Heat Exchangers	236.0	0/12	-	-
5-F-3-CHFA	Emergency Exhaust Charcoal Filter A	261.0	0/6	-	6/0
5-F-3-CHFB	Emergency Exhaust Charcoal Filter B	261.0	0/6	-	6/0
5-F-3-CHF-BAL	Emergency Exhaust Balance	261.0	-	-	2/0
4.0	Diesel Generator Building				
1-D-1-DGA-RM	Diesel Generator 1A	261.0	0/7	4/0	-
1-D-1-DGB-RM	Diesel Generator 1B	261.0	0/7	4/0	-
1-D-1-DGA-ASU	Diesel Generator Air Starting Unit 1A	261.0	1/0	-	-
1-D-1-DGB-ASU	Diesel Generator Air Starting Unit 1B	261.0	1/0	-	-
1-D-1-DGA-TK	Diesel Fuel Oil Day Tank 1A	280.0	1/0	-	-
1-D-1-DGB-TK	Diesel Fuel Oil Day Tank 1B	280.0	1/0	-	-
1-D-1-DGA-ER	Diesel Generator MCC and Control Panel 1A	261.0	-	-	1/0
1-D-1-DGB-ER	Diesel Generator MCC and Control Panel 1B	261.0	-	-	1/0
1-D-3-DGA-ES	Diesel Exhaust Silencer 1A	292.0	-	1/0	-
1-D-3-DGB-ES	Diesel Exhaust Silencer 1B	292.0	-	1/0	-

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INSERT H - INSTRUMENTATION (Cont.d)

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

Zone	Instrument Location	Elevation FT	-Minimum Number of Instruments Operable Heat (A/B)	Flame (A/B)	Smoke (A/B)
5.0	Diesel Oil Storage Tank Area				
1-0-PA	Diesel Fuel Oil Pump Room 1A	242.25	0/1	1/0	-
1-0-PB	Diesel Fuel Oil Pump Room 1B	242.25	0/1	1/0	-
5-0-BAL	Diesel Fuel Oil Storage Tank Area - Balance	242.25	-	4/0	-
6.0	Emergency Service Water Intake Structure				
12-I-ESWPA	Electrical Equipment Room SA	251.7/262.0	-	-	7/0
	Pump Room SA	262.0	-	2/0	-
12-I-ESWPB	Electrical Equipment Room SB	251.7/262.0	-	-	7/0
	Pump Room SB	262.0	-	2/0	-

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~~LOOSE-PART DETECTION SYSTEM~~ METAL IMPACT MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.9 The ~~Loose-Part Detection~~ Metal Impact Monitoring System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more ~~Loose-Part Detection~~ Metal Impact Monitoring System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each channel of the ~~Loose-Part Detection~~ Metal Impact Monitoring Systems shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ~~ANALOG~~ CHANNEL OPERATIONAL TEST, EXCEPT FOR VERIFICATION OF THE SETPOINT at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

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RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

(1)

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or (2) declare the channel inoperable AND TAKE ACTION AS DIRECTED BY b. BELOW.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.~~4~~7 why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4, are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ~~ANALOG~~ CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-8.

REFER TO NEXT 2 PAGES
FOR TABLE 3.3-12

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Line	1	35
b. Steam Generator Blowdown Effluent Line	1	36
c. Turbine Building (Floor Drains) Sumps Effluent Line	1	36
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Service Water System Effluent Line	1	37
b. Component Cooling Water System Effluent Line	1	37
3. Continuous Composite Samplers and Sampler Flow Monitor		
a. Steam Generator Blowdown Effluent Line (alternate to Item 1.b.)	1	36
b. Turbine Building Sumps Effluent Line (alternate to Item 1.c.)	1	36
4. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	1	38
b. Steam Generator Blowdown Effluent Line	1	38
c. Discharge Canal	1	38

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TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Lines		
1. Treated Laundry and Hot Shower Tanks Discharge Monitor	1	35
2. Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge Monitor	1	35
3. Secondary Waste Sample Tanks Monitor	1	35
b. Turbine Building Floor Drains Effluent Line	1	36
c. Outdoor Tank Area Drain Transfer Pump Monitor	1	37
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Normal Service Water System Return From Waste Processing Building to the Circulating Water System	1	37
b. Normal Service Water System Return From the Reactor Auxiliary Building to the Circulating Water System	1	37

TABLE 3.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Lines		
1. Treated Laundry and Hot Shower Tanks Discharge.	1	38
2. Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge	1	38
3. Secondary Waste Sample Tank	1	38
4. Normal Service Water System Return From Waste Processing Building to the Circulating Water System	1	38
5. Normal Service Water System Return From the Reactor Auxiliary Building to the Circulating Water System	1	38

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TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for radioactivity for up to 30 days at a lower limit of detection of no more than 10^{-7} microCurie/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microCurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microCurie/gram DOSE EQUIVALENT I-131.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

~~ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the radioactivity level is determined at least once per 4 hours during actual releases.~~

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REFER TO TABLE 4.3-8 INSERT

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluent Line	D	P	R(3)	Q(1)
b. Steam Generator Blowdown Effluent Line	D	M	R(3)	Q(1)
c. Turbine Building (Floor Drains) Sumps Effluent Line	D	M	R(3)	Q(1)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Service Water System Effluent Line	D	M	R(3)	Q(2)
b. Component Cooling Water System Effluent Line	D	M	R(3)	Q(2)
3. Continuous Composite Samplers and Sampler Flow Monitor				
a. Steam Generator Blowdown Effluent Line (alternate to Item 1.b.)	D	N.A.	R	Q
b. Turbine Building Sumps Effluent Line (alternate to Item 1.c.)	D	N.A.	R	Q

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REFER TO TABLE 4.3-8 INSERT

TABLE 4.3-8 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST
4. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line	D(4)	N.A.	R	Q
c. Discharge Canal	D(4)	N.A.	R	Q
5. Radioactivity Recorders*				
a. Liquid Radwaste Effluent Line	D	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line	D	N.A.	R	Q

*Required only if Alarm/Trip Setpoint in based on recorder-controller.

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TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluent Lines				
1. Treated Laundry and Hot Shower Tanks Discharge Monitor	D	P	R(3)	Q(1)
2. Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge Monitor.	D	P	R(3)	Q(1)
3. Secondary Waste Sample Tank Discharge Monitor	D	P	R(3)	Q(1)
b. Turbine Building Floor Drains Effluent Line	D	M	R(3)	Q(1)
c. Outdoor Tank Area Drain Transfer Pump Monitor	D	M	R(3)	Q(2)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Normal Service Water System Return From the Waste Processing Building to the Circulating Water System	D	M	R(3)	Q(2)

TABLE 4.3-8 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST
b. Normal Service Water System Return From the Reactor Auxiliary Building to the Circulating Water System	D	M	R(3)	Q(2)
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Lines	D(4)	N.A.	R	Q
1. Treated Laundry and Hot Shower Tanks Discharge	D(4)	N.A.	R	Q
2. Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge	D(4)	N.A.	R	Q
3. Secondary Waste Sample Tank	D(4)	N.A.	R	Q
4. Normal Service Water System Return From Waste Processing Building to the Circulating Water System	D(4)	N.A.	R	Q
5. Normal Service Water System Return From Reactor Auxiliary Building to the Circulating Water System	D(4)	N.A.	R	Q

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TABLE 4.3-8 (Continued)

TABLE NOTATIONS

(1) The ~~ANALOG~~ CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Power ~~Circuit~~ failure, ~~or~~

~~c. Instrument indicates a downscale failure, or~~

~~d. Instrument controls not set in operate mode.~~

(2) The ~~ANALOG~~ CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Power ~~Circuit~~ failure, ~~or~~

~~c. Instrument indicates a downscale failure, or~~

d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

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RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent (1) monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or (2) declare the channel inoperable AND TAKE ACTION AS DIRECTED BY b. BELOW.

b. With the number of OPERABLE radioactive gaseous effluent monitoring instrumentation channels less than the Minimum Channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ~~ANALOG~~ CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

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TABLE 3.3-13

REFER TO FOLLOWING PAGES FOR TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	45
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Effluent System Flow Rate Measuring Device	1	*	46
e. Sampler Flow Rate Measuring Device	1	*	46
2A. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitor (Automatic Control)	1	**	49
b. Hydrogen or Oxygen Monitor (Process)	1	**	49

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TABLE 3.3-11

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GASEOUS WASTE PROCESSING SYSTEM-HYDROGEN AND OXYGEN ANALYZERS			
a. Hydrogen Monitor~~s~~	2 ~~2~~/recombiner	**	49 ~~50, 52~~
b. Oxygen Monitors	2/recombiner	**	50, 52
c. Oxygen Monitor****	1/compressor	***	49 ~~50, 53~~
2. TURBINE BUILDING VENT STACK			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler#	1	*	51
c. Particulate Sampler#	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor#	1	*	46
3. PLANT VENT STACK			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler#	1	*	51
c. Particulate Sampler#	1	*	51
d. Flow Rate Monitor	1	*	46

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
	e. Sampler Flow Rate Monitor#	1	*	46
4.	WASTE PROCESSING BUILDING VENT STACK 5			
	a. Nob.e Gas Activity Monitor	1	*	45, 47
	b. Iodine Sampler#	1	*	51
	c. Particulate Sampler#	1	*	51
	d. Flow Rate Monitor	1	*	46
	e. Sampler Flow Rate Monitor#	1	*	46
5.	WASTE PROCESSING BUILDING STACK 5A			
	a. Noble Gas Activity Monitor	1	*	47
	b. Iodine Sampler#	1	*	51
	c. Particulate Sampler#	1	*	51
	d. Flow Rate Monitor	1	*	46
	e. Sampler Flow Rate Monitor#	1	*	46

- These samplers are located on the High Range Monitor skid for this release point.

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
2B. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems not designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitors (Automatic control, redundant)	2	**	50, 52
b. Hydrogen or Oxygen Monitors (Process, dual)	2	**	50
3. Condenser Evacuation System			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. Vent Header System			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46
5. Containment Purge System			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	48
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
6. Auxiliary Building Ventilation System			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46
7. Fuel Storage Area Ventilation System			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
8. Radwaste Area Ventilation System			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46
9. Other Exhaust and Vent Systems such as:			
. Steam Generator Blowdown Vent System and Turbine Gland Seal Condenser Exhaust			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46

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TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE TREATMENT ~~WASTE GAS HOLDUP~~ SYSTEM operation.

ACTION STATEMENTS

ACTION 45 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

~~ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.~~

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this ~~WASTE GAS HOLDUP~~ GASEOUS RADWASTE TREATMENT SYSTEM may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 50 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both channels inoperable, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

*** DURING GASEOUS RADWASTE TREATMENT SYSTEM OPERATION IN THE HIGH PRESSURE MODE.

**** IF OPERABLE, THIS MONITOR MAY BE USED TO SATISFY THE REQUIREMENTS OF ITEM 1.b.

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TABLE 3.3-13 (Continued)

TABLE NOTATIONS (Continued)

ACTION 51 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 52 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

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TABLE 4.3-9 REFER TO FOLLOWING PAGES FOR TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Measuring Device	P	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
1. ~~2A. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems designed to withstand the effects of a hydrogen explosion)~~ GASEOUS WASTE PROCESSING SYSTEM- HYDROGEN AND OXYGEN ANALYZERS					
a. Hydrogen Monitor ~~(Automatic Control)~~	D	N.A.	Q(4)	M	**
b. ~~Hydrogen or~~ Oxygen Monitor ~~(Process)~~	D	N.A.	~~Q(4) or~~ Q(5)	M	**
c. OXYGEN MONITOR	D	N.A.	Q(5)	M	***

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TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GASEOUS WASTE PROCESSING SYSTEM-HYDROGEN AND OXYGEN ANALYZERS					
a. Hydrogen Monitors	D	N.A.	Q(4)	M	**
b. Oxygen Monitors	D	N.A.	Q(5)	M	**
c. Oxygen Monitor	D	N.A.	Q(5)	M	***
2. TURBINE BUILDING STACK					
a. Noble Gas Activity	D	M	R(3)	Q(2)	*
b. Iodine Sampler#	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler#	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor#	D	N.A.	R	Q	*
3. PLANT VENT STACK					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler#	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler#	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
e. Sampler Flow Rate Monitor#	D	N.A.	R	Q	*
4. WASTE PROCESSING BUILDING VENT STACK 5					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler#	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler#	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor#	D	N.A.	R	Q	*
5. WASTE PROCESSING BUILDING VENT STOCK 5A					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler#	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler#	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor#	D	N.A.	R	Q	*

These samplers are located on the High Range Monitor skid for this release point.

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~~W-STS~~ SHEARON HARRIS UNIT 1

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
~~10. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems not designed to withstand the effects of a hydrogen explosion)~~					
~~a. Hydrogen Monitors (Automatic Control, redundant)~~	~~D~~	~~N.A.~~	~~Q(4)~~	~~M~~	~~**~~
~~b. Hydrogen or Oxygen Monitors (Process, dual)~~	~~D~~	~~N.A.~~	~~Q(4) or Q(5)~~	~~M~~	~~**~~
2~~1~~. TURBINE BUILDING STACK ~~Condenser Evacuation System~~					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler #	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler #	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor #	D	N.A.	R	Q	*
3~~K~~. PLANT VENT STACK ~~Vent Header System~~					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler #	W	N.A.	N.A.	N.A.	*

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TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3 ~~A.~~ PLANT VENT STACK ~~Vent Header System~~ (Continued)					
c. Particulate Sampler #	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor #	D	N.A.	R	Q	*
4 ~~B.~~ WASTE PROCESSING BUILDING VENT STACK 5 ~~Containment Purge System~~					
a. Noble Gas Activity Monitor - ~~Providing Alarm and Automatic Termination of Release~~	D	M ~~X~~	R(3)	Q(1)	*
b. Iodine Sampler #	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler #	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor #	D	N.A.	R	Q	*
5 ~~C.~~ WASTE PROCESSING BUILDING VENT STACK 5A ~~Auxiliary Building Ventilation System~~					
a. Noble Gas Actvity Monitor	D	M	R(3)	Q(2)	*

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~~#3.5~~ SHEARON HARRIS UNIT 1

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	~~ANALOG~~ CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5~~6~~. WASTE PROCESSING BUILDING VENT STACK 5A (CONTINUED) ~~Auxiliary Building Ventilation System (Continued)~~					
b. Iodine Sampler #	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler #	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor #	D	N.A.	R	Q	*
6~~7~~. CONTAINMENT PRE-ENTRY PURGE ISOLATION ~~Fuel Storage Area Ventilation System~~					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	* *
~~b. Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
c. Particulate ~~Sampler~~ MONITOR	W	N.A.	N.A.	N.A.	* *
~~d. Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~

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TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Radwaste Area Ventilation System					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
9. Other Exhaust and Vent Systems such as:					
Steam Generator Blowdown Vent System, Turbine Gland Seal Condenser Exhaust					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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~~# THESE SAMPLERS ARE LOCATED ON THE High Range Monitor SKID FOR THIS RELEASE POINT.~~

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TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* At all times.

** During ~~WASTE GAS HOLDUP SYSTEM~~ GASEOUS RADWASTE TREATMENT SYSTEM operation.

*** DURING GASEOUS RADWASTE TREATMENT SYSTEM OPERATION IN THE HIGH PRESSURE MODE.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. ~~Circuit~~ POWER failure, ~~or~~

~~c. Instrument indicates a downscale failure, or~~

c ~~d~~. Instrument controls not set in operate mode.

(2) The ~~ANALOG~~ CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. ~~Circuit~~ POWER failure, ~~or~~ OR

~~c. Instrument indicates a downscale failure, or~~

c ~~d~~. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

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3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one ~~stop~~ THROTTLE valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours. The provisions of Specification 3.0.4 are not applicable to this ACTION.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position:

1) ~~{Four}~~ high pressure turbine ~~stop~~ THROTTLE valves,

2) ~~{Four}~~ high pressure turbine governor valves,

3) ~~{Four}~~ low pressure turbine reheat stop valves, and

4) ~~{Four}~~ low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position,

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks, and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

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* Not applicable in Mode 2 or 3 with all main steam isolation valves and bypass valves in the closed position and all other steam flow paths to the turbine isolated.

Capitalize the Titles of the Following System and Component names:

Reactor Coolant System

Reactor Coolant Loop

Residual Heat Removal Loop

Reactor Coolant Loops

Steam Generator

Reactor Coolant Pump

Power Operated Relief Values

Containment Sump

Reactor Head Flange Leakoff

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

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HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant loops in operation when the Reactor Trip System breakers are closed and one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop ~~[A]~~ and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop ~~[B]~~ and its associated steam generator and reactor coolant pump, and

c. Reactor Coolant Loop ~~[C]~~ and its associated steam generator and reactor coolant pump. ~~and~~

~~d. Reactor Coolant Loop [D] and its associated steam generator and reactor coolant pump.~~

APPLICABILITY: MODE 3.**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to ~~[17%]~~ 10% narrow range at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**See Special Test Exception Specification 3.10.4

~~* STS~~

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REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop ~~[~~A~~]~~ and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop ~~[~~B~~]~~ and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop ~~[~~C~~]~~ and its associated steam generator and reactor coolant pump,**

~~d. Reactor Coolant Loop [D] and its associated steam generator and reactor coolant pump,**~~

d ~~e~~. RHR Loop ~~[~~A~~]~~, and

e ~~f~~. RHR Loop ~~[~~B~~]~~.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

---

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 250 ~~[275]~~°F unless the secondary water temperature of each steam generator is less than 50 °F above each of the Reactor Coolant System cold leg temperatures.

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SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to ~~[17]%~~ 10% at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

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REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The secondary side water level of at least two steam generators shall be greater than ~~[17]~~ 10 % OF THE NARROW RANGE LEVEL INDICATORS.

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to ~~[275]~~ 250°F unless the secondary water temperature of each steam generator is less than 50 °F above each of the Reactor Coolant System cold leg temperatures.

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REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

---

*One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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REACTOR COOLANT SYSTEM

ISOLATED LOOP (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.4.1.5 The boron concentration of an isolated loop shall be maintained greater than or equal to the boron concentration of the operating loops.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

With the requirements of the above specification not satisfied, do not open the isolated loop's stop valves; either increase the boron concentration of the isolated loop to within the limits within 4 hours or be in at least HOT STANDBY within the next 6 hours with the unisolated portion of the RCS borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F.

SURVEILLANCE REQUIREMENTS

4.4.1.5 The boron concentration of an isolated loop shall be determined to be greater than or equal to the boron concentration of the operating loops at least once per 24 hours and within 30 minutes prior to opening either the hot leg or cold leg stop valves of an isolated loop.

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REACTOR COOLANT SYSTEM

ISOLATED LOOP STARTUP [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated until:

a. The isolated loop has been operating on a recirculation flow of greater than or equal to ___ gpm for at least 90 minutes and the temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops, and

b. The reactor is subcritical by at least 1% $\Delta k/k$.

APPLICABILITY: All MODES.

ACTION:

With the requirements of the above specification not satisfied, suspend startup of the isolated loop.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.2 The reactor shall be determined to be subcritical by at least 1% $\Delta k/k$ within 30 minutes prior to opening the cold leg stop valve.

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REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immedi:tely suspend all operations involving positive reactivity changes and p ace an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

1227 ≤ EQUIVALENT TO 92% OF INDICATED SPAN

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to ~~[1656]~~ cubic feet, and at least two groups of pressurizer heaters each having a capacity of at least ~~[150]~~ 125 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

~~4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.~~

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With ~~one~~ TWO PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY wtihin the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With ~~both~~ ~~MORE THAN ONE~~ ALL 3 PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its assoicated solenoid valve; and (2) apply the ACTION b. or c. above, as appropriate, for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

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3/4.4.4 RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION a.) b. or c. in Specification 3.4.4.

4.4.4.3 ~~The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:~~

~~a. Manually transferring motive and control power from the normal to the emergency power supply, and~~

~~b. Operating the valves through a complete cycle of full travel.~~

THE BACKUP ACCUMULATORS FOR THE PORV's SHALL BE DEMONSTRATED OPERABLE AT LEAST ONCE PER 18 MONTHS BY ISOLATING THE NORMAL AIR SUPPLY AND OPERATING THE VALVES THROUGH A COMPLETE CYCLE OF FULL TRAVEL.

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REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2 A or B. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection ~~(subsequent to the preservice inspection)~~ of each steam generator shall include:

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1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes A or B selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

INSERT

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

d. EACH INSPECTION SHALL INCLUDE A SAMPLE OF THOSE TUBES EXPANDED IN THE PREHEATER SECTION OF THE STEAM GENERATOR. THE FIRST SAMPLE SIZE, SECOND SAMPLE SIZE AND SUBSEQUENT INSPECTION SHALL FOLLOW TABLE 4.4-2B.

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4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection A or B of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled A or B inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to ~~[40]%* [later]~~ 40% of the nominal tube wall thickness ~~FOR TUBES EXPANDED IN THE PREHEATER SECTION AND 40% OF THE NOMINAL TUBE WALL THICKNESS FOR ALL OTHER TUBES~~;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

~~*Value to be determined in accordance with the recommendations of Regulatory Guide 1.121, August 1976.~~

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~~9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.~~

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2 Aor B.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

~~Preservice Inspection~~	~~No~~			~~Yes~~		
No. of Steam Generators per Unit	~~Two~~	Three	~~Four~~	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 ~~N~~ % of the tubes ~~(where N is the number of steam generators in the plant)~~ if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

~~3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.~~

TABLE 4.4-2A

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. ~~Prompt~~ Notification to NRC pursuant to specification 6.9.1.2	All other S. G.s are C-1	None	N/A	N/A
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. ~~Prompt~~ Notification to NRC pursuant to specification 6.9.1.2	N/A	N/A

$S = \cancel{3}\,\frac{9\,\cancel{N}}{n}\%$ Where ~~N is the number of steam generators in the unit, and~~ n is the number of steam generators inspected during an inspection.

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TABLE 4.4-2B

STEAM GENERATOR TUBE INSPECTION - TUBES EXPANDED IN PREHEATER REGION

1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of S of the tubes expanded in the preheater section	C-1	None	N/A	N/A
	C-2	plug defective tubes and inspect all other expanded tubes in this Steam Generator	C-1	N/A
			C-2	Plug defective tubes
			C-3	Perform action for C-3 result of first sample.
	C-3	Inspect all expanded tubes in this Steam Generator, plug defective tubes and inspect all expanded tubes in each other Steam Generator Notification to NRC pursuant to Specification 6.9.2	All other SP's are C-1	None
			One or more S.G.'s C-2 but no additional SG are C-3	Plug defective tubes
			Additional SG is C-3	Plug defective tubes, Notification to NRC pursuant to specification 6.9.2.

$S = \frac{9}{n}$ % where n is the number of steam generators inspected during an inspection.

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3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere ~~[~~Gaseous ~~or Particulate]~~ Radioactivity Monitoring System,

b. The Containment ~~Pocket~~ Sump Level and Flow Monitoring System, and

c. ~~Either~~ The ~~[containment air cooler condensate flow rate] or a~~ Containment Atmosphere ~~[Gaseous or~~ Particulate~~]~~ Radioactivity Monitoring System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only two of the above required Leakage Detection Systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required Gaseous or Particulate Radioactive Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring Systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ~~ANALOG~~ CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

~~b. Containment Pocket Sump Level and Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months, and~~

~~c. [Specify appropriate surveillance tests depending upon the type of Leakage Detection System selected.]~~

b. PERFORMANCE OF A CHANNEL CALIBRATION of THE FOLLOWING INSTRUMENTS AT LEAST ONCE PER 18 MONTHS:
1. CONTAINMENT SUMP LEVEL
2. LEAKAGE FLOW MONITORING

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REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators ~~not isolated from the Reactor Coolant System~~ and ~~[500]~~ gallons per day through any one steam generator, ~~not isolated from the Reactor Coolant System~~,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 31 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

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REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere [gaseous or particulate] radioactivity monitor at least once per 12 hours;

b. Monitoring the containment ~~pocket~~ sump inventory and discharge at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

~~a. At least once per 18 months,~~

~~b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,~~

~~c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and~~

~~d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.~~

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

a. FOR CHECK VALVES:
1. IF THE VALVE HAS BEEN DISTURBED BECAUSE OF FLOW IN THE LINE, OR
2. AT LEAST ONCE EVERY EIGTHEEN MONTHS; OR
3. FOLLOWING MAINTENANCE, REPAIR, OR REPLACEMENT WORK.

b. FOR MOTOR OPERATED VALVES:
1. AT LEAST EVERY EIGHTEEN MONTHS; OR
2. FOLLOWING MAINTENANCE, REPAIR, OR REPLACEMENT WORK.

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TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	TYPE	FUNCTION
1 RH 1	MOV	RHR PUMP SUCTION
1 RH 2	MOV	RHR PUMP SUCTION
1 RH 39	MOV	RHR PUMP SUCTION
1 RH 40	MOV	RHR PUMP SUCTION
1 SI 134	CHECK	LOW HEAD INJECTION (HOT LEG)
1 SI 135	CHECK	LOW HEAD INJECTION (HOT LEG)
1 SI 249	CHECK	ACCUMULATOR INJECTION
1 SI 250	CHECK	ACCUMULATOR INJECTION
1 SI 251	CHECK	ACCUMULATOR INJECTION
1 SI 252	CHECK	ACCUMULATOR INJECTION
1 SI 253	CHECK	ACCUMULATOR INJECTION
1 SI 254	CHECK	ACCUMULATOR INJECTION
1 SI 346	CHECK	LOW HEAD INJECTION
1 SI 347	CHECK	LOW HEAD INJECTION
1 SI 356	CHECK	LOW HEAD INJECTION
1 SI 357	CHECK	LOW HEAD INJECTION
1 SI 358	CHECK	LOW HEAD INJECTION

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REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

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TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	< 0.10 ppm	≤ 1.00 ppm
Chloride	< 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

*Limit not applicable with T_{avg} less than or equal to 250°F.

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TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least once per 72 hours
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

*Not required with T_{avg} less than or equal to 250°F

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REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microCuries per gram of gross radioactivity

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or for more than 800 hours in any consecutive 12-month period, or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours~~; and~~ the provisions of specification 3.0. are not applicable;

b. With the gross specific activity of the reactor coolant greater than 100/$\bar{E}$ microCuries per gram of gross radioactivity, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ micro-Curies per gram of gross radioactivity, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

* With Tavg greater than or equal to 500°F.

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SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

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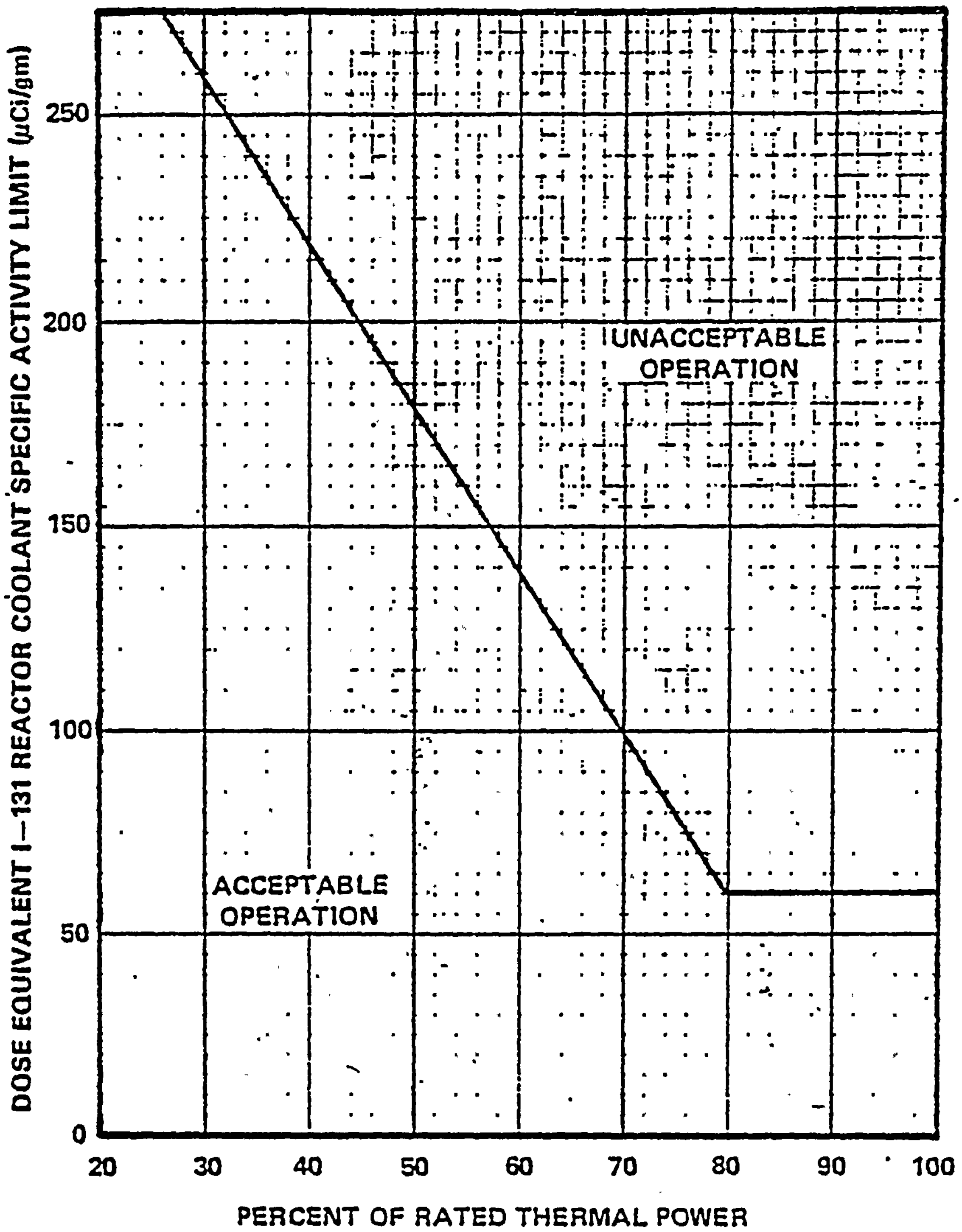

FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 µCi/gram DOSE EQUIVALENT I-131

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TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination*	At least once per 72 hours.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
3. Radiochemical for Ē Determination**	1 per 6 months***	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 µCi/gram DOSE EQUIVALENT I-131 or 100/Ē µCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

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TABLE 4.4-4 (Continued)

TABLE NOTATIONS

*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.

**~~A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radio-nuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level.~~

***Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the Reactor Coolant System is restored within its limits.

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of [100]°F in any 1-hour period,

b. A maximum cooldown of [100]°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to [10]°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: ~~At all times.~~ MODES 1, 2, 3 AND 4

ACTION:

OR INSPECTION

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor ~~Coolant System;~~ determine that the Reactor ~~Coolant System~~ remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

PRESSURE VESSEL

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.~~1~~ 2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of ~~[100]~~ (LATER) °F in any 1-hour period,

b. A maximum cooldown of ~~[100]~~ (LATER) °F in any 1-hour period, and

c. A maximum temperature change of less than or equal to [10]°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: ~~At all times~~. Mode 5 and 6

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure (OR INSPECTION) to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor ~~Coolant System~~ Vessel; determine that the Reactor ~~Coolant System~~ Vessel remains acceptable for continued operation or ~~be in at least HOT STANDBY within the next 6 hours and reduce~~ maintain the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.~~1~~ 2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

~~4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.~~

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MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	:	PLATE METAL
COPPER CONTENT	:	0.10 WT%
PHOSPHORUS CONTENT	:	0.006 WT%
RT_{NDT} INITIAL	:	90°F
RT_{NDT} AFTER 5 EFPY	:	1/4T, 155°F 3/4T, 135°F

AND 100°F/HR

CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 5 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

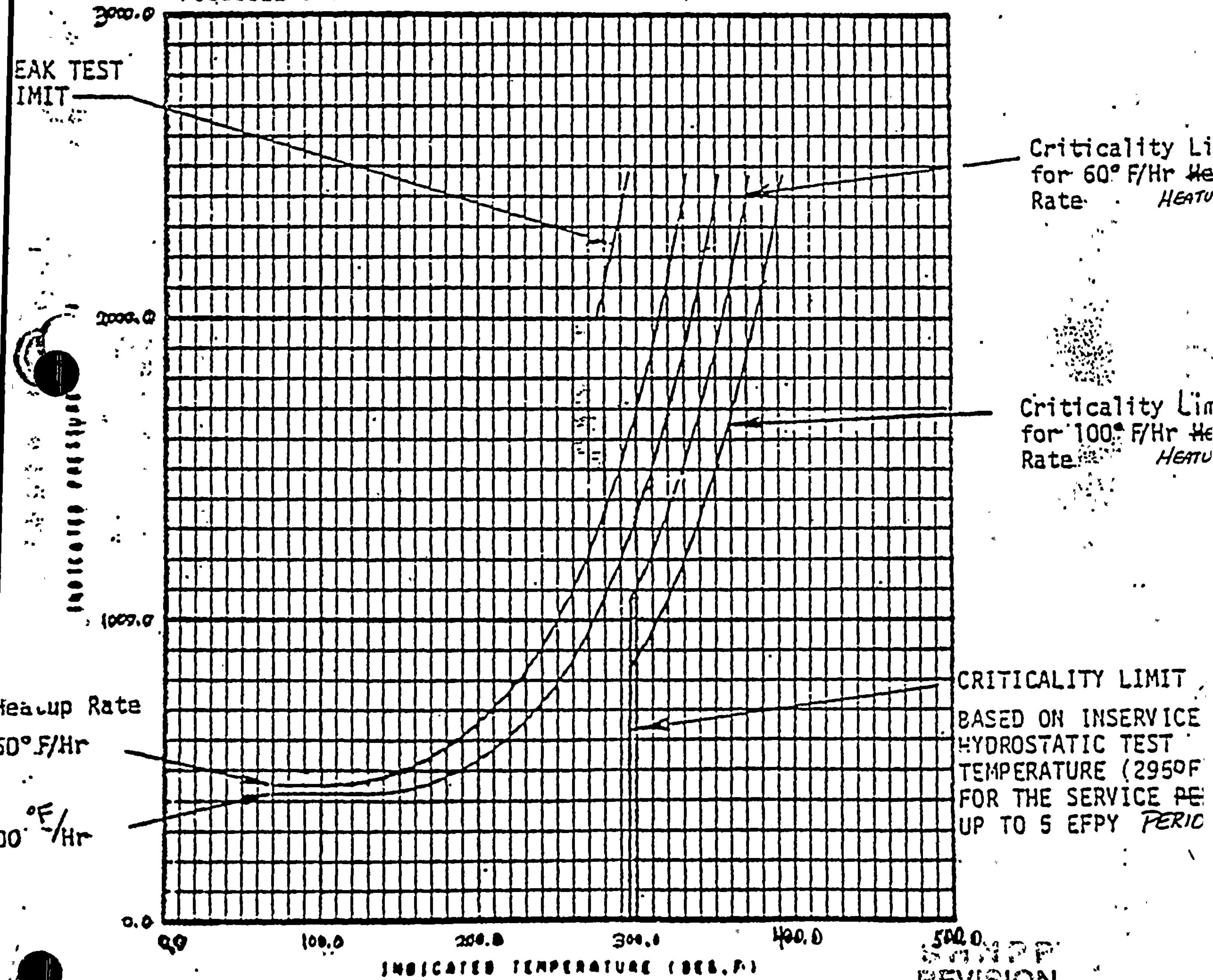

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FIGURE 3.4-2 SHEARON HARRIS UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE FOR 5 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL : PLATE METAL
COPPER CONTENT : 0.10 WT%
PHOPHORUS CONTENT : 0.006WT%
RT_{NDT} INITIAL : 90°F
RT_{NDT} AFTER 5 EFPY : 1/4T, 155°F
3/4T, 135°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 5 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

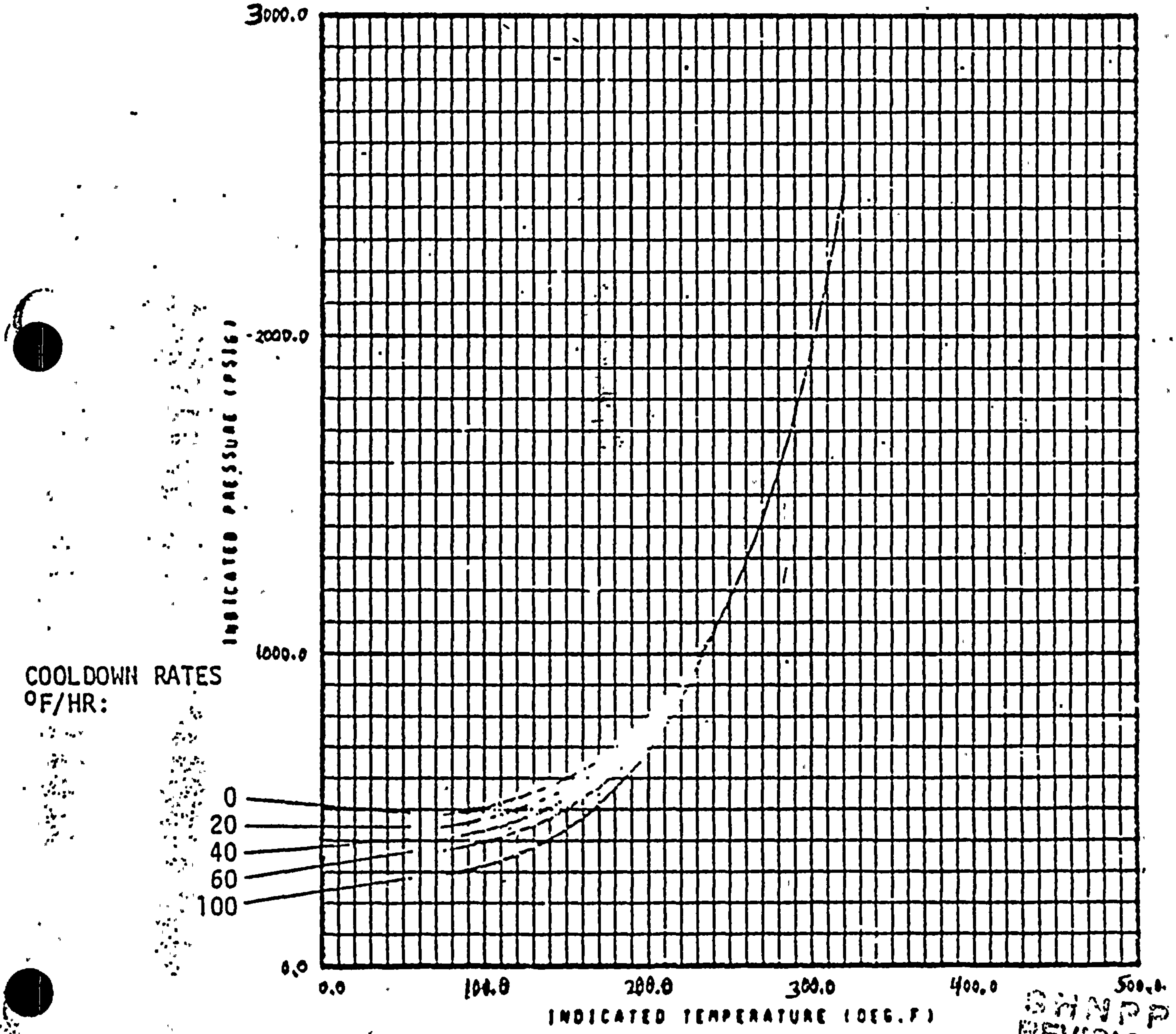

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FIGURE ~~1~~ - SHEARON HARRIS UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE FOR 5 EFPY

SHEARON HARRIS UNIT 1

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~~SHNPP~~ SHEARON HARRIS UNIT 1

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD * FACTOR	WITHDRAWAL TIME (EFPY)
U	343°	3.12	1ST REFUELING
V	107°	3.12	3 EFPY
X	287°	3.12	6 EFPY
W	110°	2.7	12 EFPY
Y	290°	2.7	20 EFPY
Z	340°	2.7	STANDBY

* THE FACTOR BY WHICH THE CAPSULE FLUENCE LEADS THE VESSEL MAXIMUM INNER WALL FLUENCE.

** Withdrawal time may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedules.

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REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.~~2~~ 3 The pressurizer temperature shall be limited to:

a. A maximum heatup of ~~[~~100~~]~~°F in any 1-hour period,

b. A maximum cooldown of ~~[~~200~~]~~°F in any 1-hour period, and

c. A maximum spray water temperature differential of ~~[320]~~ 625°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.~~2~~ 3 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.~~3~~ 4 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with ~~a lift setting of less than or equal to [450] psig, or~~ SETPOINTS WHICH DO NOT EXCEED THE LIMITS ESTABLISHED IN FIGURE 3.4-4, OR

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.45 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to ~~[275]~~ ~~350~~ LATER °F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.45 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.45 square inch vent within 8 hours.

c. In the event either the PORVs or the RCS vent(s) ~~established pursuant to ACTION~~ are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

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OVERPRESSURE PROTECTION SYSTEM

SURVEILLANCE REQUIREMENTS

4

4.4.9.~~3~~.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ~~ANALOG~~ CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4

4.4.9.~~3~~.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE SYSTEM.

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REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

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REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System Vent path consisting of at least one ~~[two]~~ vent valves(s) and ~~[~~one~~]~~ block valve powered from emergency busses shall be OPERABLE and closed at each of the following locations:

a. ~~[~~Reactor vessel head~~]~~,

b. ~~[~~Pressurizer steam space~~]~~~~, and~~

~~c. [Reactor Coolant System high point].~~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and both block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two ~~or more~~ Reactor Coolant System vent paths inoperable due to causes other than the removal of power to both block valves pursuant to ACTION a, maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one ~~[two]~~ of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 is not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each Reactor Coolant System vent path block valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel from the control room.

4.4.11.2 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

~~a. Verifying all manual isolation valves in each vent path are locked in the open position~~,

a ~~b~~. Cycling each vent valve through at least one complete cycle of full travel from the control room, and

b ~~c~~. Verifying flow through the Reactor Coolant System vent paths during venting.

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Capitalize the titles of the following Systems and Component Names:

Reactor Coolant System Accumulator

Accumulator

Use "Charging/Safety Injection Pump" for "charging pump."

Refueling Water Storage Tank

Residual Heat Removal Pump

Residual Heat Removal Heat Exchanger

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between [6190] and [6560] gallons,

c. A boron concentration of between [1900] and [2100] ppm, and

d. A nitrogen cover-pressure of between [603] and [686] psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to [1% of tank volume] by verifying the boron concentration of the accumulator solution; and

*Pressurizer pressure above 1000 psig.

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 31 days when the RCS pressure is above [1000 psig] by verifying that power to the isolation valve operator is disconnected by removal of the breaker from the circuit.

[d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:

1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

2) Upon receipt of a Safety Injection test signal.]

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between ~~[6190]~~ 7330 and ~~[6560]~~ 7810 gallons which is equivalent to an indicated level of between 66 and 96% level.

c. A boron concentration of between [1900] and ~~[2100]~~ ppm, and

d. A nitrogen cover-pressure of between ~~[603]~~ 585 and ~~[686]~~ 665 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to ~~[1% of tank volume]~~ 69 GALLONS WHICH IS EQUIVALENT TO AN INDICATED LEVEL CHANGE OF 8% by verifying the boron concentration of the accumulator solution; and

c. At least once per 31 days when the RCS pressure is above [1000 psig] by verifying that ~~power to the isolation valve operator is disconnected by removal of the breaker from the circuit.~~ THE CIRCUIT BREAKER SUPPLYING POWER TO THE RESPECTIVE ISOLATION VALVE OPERATOR IS OPEN.

~~[d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:~~

~~1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and~~

~~2) Upon receipt of a Safety Injection test signal.]~~

4.5.1.1.2 Each Cold Leg Injection Accumulator System water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ~~ANALOG~~ CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

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3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,#

~~b. One OPERABLE Safety Injection pump (four loop plants only),~~

b ~~c~~. One OPERABLE RHR heat exchanger,

c ~~d~~. One OPERABLE RHR pump, and

d ~~e~~. An OPERABLE flow path capable of taking suction from upon being manually aligned the refueling water storage tank on a Safety Injection signal and ~~automatically~~ manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

A MAXIMUM OF TWO CHARGING/SAFETY INJECTION PUMPS SHALL BE OPERABLE WHENEVER THE TEMPERATURE OF ALL THREE OF THE RCS COLD LEGS IS GREATER THAN 250°F.

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EMERGENCY CORE COOLING SYSTEMS

UPPER HEAD INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each Upper Head Injection Accumulator System shall be OPERABLE with:

a. The isolation valves open,

b. The water-filled accumulator containing a minimum of [1850] cubic feet of borated water having a boron concentration of between [1900] and [2100] ppm, and

c. The nitrogen-bearing accumulator pressurized to between ____ and ____ psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With the Upper Head Injection Accumulator System inoperable, except as a result of a closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1900 psig within the following 6 hours.

b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1900 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.2 Each Upper Head Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the containment borated water volume and nitrogen pressure in a accumulator, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1900 psig.

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4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed AS NOTED: ~~BELO~~

~~Valve Number~~	~~Valve Function~~	~~Valve Position~~

REFER TO ECCS INSERT A

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting ~~the ECCS pump casings and~~ accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to ~~[~~750~~]~~ psig the interlocks will cause the valves to automatically close.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

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b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to [1% of tank volume] by verifying the boron concentration of the solution in the water-filled accumulator;

c. At least once per 18 months by:

1) Verifying that each accumulator isolation valve closes automatically when the water level in the accumulator is [93.2 ± 2.7 inches] above the working line on the water-filled accumulator, and

2) Verifying that the total dissolved nitrogen and air in the water-filled accumulators is less than [80] scf per [1800] cubic feet of water.

d. At least once per 5 years and if the requirements of Specification 4.5.1.2c.2) are not met, by replacing the membrane installed between the water-filled and nitrogen-bearing accumulators.

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e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on (Safety Injection actuation and ~~Switcho~~ Safety Injection ~~Automatic~~ Switchover to Containment Sump) test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

~~b) Safety Injection pump, and~~

b ~~c~~) RHR pump.

f. By verifying that each of the following pumps develops the REQUIRED ~~indicated~~ differential pressure ~~on recirculation flow~~ when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump Refer to Specification 4.1.2.3.1 ~~≥ ________ psid,~~

~~2) Safety Injection pump ≥ ________ psid, and~~

3) RHR pump ~~≥ ________ psid.~~ 134 psid.

g. By verifying the correct position of each of ~~electrical and/or mechanical position stop for~~ the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE; and

2) At least once per 18 months.

HPSI System Valve Number	~~LPSI System Valve Number~~
ISI-5	
ISI-6	
ISI-7	
ISI-69	
ISI-70	
ISI-71	
ISI-101	
ISI-102	
ISI-103	
ISI-124	
ISI-125	
ISI-126	

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ECCS INSERT A

VALVE NO.	VALVE FUNCTION	VALVE POSITION-*
ISI-107	High Head Safety Injection to Reactor Coolant System Hot Legs	Closed-1
ISI-86	High Head Safety Injection to Reactor Coolant System Hot Legs	Closed-1
ISI-52	High Head Safety Injection to Reactor Coolant System Cold Legs	Closed-1
ISI-340	Low Head Safety Injection to Reactor Coolant System Cold Legs	Open-1
ISI-341	Low Head Safety Injection to Reactor Coolant System Cold Legs	Open-1
ISI-359	Low Head Safety Injection to Reactor Coolant System Hot Legs	Closed-1

* Closed-1 and Open-1 - The permissive interlock switch shall be maintained in the "OFF" position and the control switch shall be maintained in the valve position noted above

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EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 379 gpm, and

b) The total pump flow rate is less than or equal to 650 gpm.

~~2) For Safety Injection pump lines, with a single pump running:~~

~~a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to ____ gpm, and~~

~~b) The total pump flow rate is less than or equal to ____ gpm.~~

2 ~~3~~) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3663 gpm.

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3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,*

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 ~~20~~ hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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* A maximum of one centrifugal charging pump ~~and one Safety Injection pump~~ shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to ~~[275]~~ 250°F.

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EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 All charging pumps ~~and Safety Injection pumps~~, except the above allowed OPERABLE pump~~s~~, shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position ~~at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to [275]°F.~~ WITHIN 4 HOURS ~~WITHIN 4 HOURS~~ AFTER ~~ENTERING MODE 4 FROM MODE 3~~ PRIOR TO THE TEMPERATURE OF ONE OR MORE OF THE RCS COLD LEGS DECREASING BELOW 225 ~~250~~°F AND AT LEAST ONCE PER 31 DAYS THEREAFTER.

ONE OR MORE OF THE RCS COLD LEGS DECREASES BELOW 250°F

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3/4.5.~~5~~ 4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.~~5~~ 4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of ~~422,000~~ LATER gallons,

b. A minimum boron concentration of [2000] ppm of boron,

c. A minimum solution temperature of ~~[35]~~ 40 °F, and

d. A maximum solution temperature of ~~[100]~~ LATER °F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.~~5~~ 4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the [outside] air temperature is less than ~~[35]~~ 40 °F or greater than ~~[100]~~ LATER °F.

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EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A contained borated water volume of between ______ and ______ gallons,

b. A boron concentration of between 20,000 and 22,500 ppm, and

c. A minimum solution temperature of 145°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% Δk/k at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days,

b. Verifying the boron concentration of the water in the tank at least once per 7 days, and

c. Verifying the water temperature at least once per 24 hours.

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EMERGENCY CORE COOLING SYSTEMS

HEAT TRACING [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one channel of heat tracing on either the boron injection tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to [145]°F at least once per 8 hours; otherwise, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 24 hours by verifying the tank and flow path temperatures to be greater than or equal to [145]°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

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SECTION 3/4.6A

CONTAINMENT SYSTEMS SPECIFICATIONS

FOR

WESTINGHOUSE

ATMOSPHERIC TYPE CONTAINMENT

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Capitalize the titles of the following systems and component names:

Personnel Air Locks
Preentry Purge Makeup and Exhaust
Normal Containment Purge Makeup and Exhaust
Containment Spray System
Spray Additive Tank
Containment Fan Coolers
Containment Hydrogen Monitors
Containment Hydrogen Recombiners

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their CLOSED positions, except as provided in ~~Table 3.6-1 of~~ Specification 3.6.4.1;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 41 psig ~~[50 psig]~~, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

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CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, .10 ~~[0.20]~~% by weight of the containment air per 24 hours at P_a, 41 ~~[50~~ psig~~]~~, or

2) Less than or equal to L_t, 0.07 ~~[0.14]~~% by weight of the containment air per 24 hours at a reduced pressure of P_t, 20.5 ~~[25~~ psig~~]~~.

b. A combined leakage rate of less than OR EQUAL TO 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than 0.75 L_t, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-~~[1972]~~. A SHORT DURATION TEST MAY BE PERFORMED FOR TYPE A TESTS. USING THE TEST DURATION CRITERIA CONTAINED IN PARAGRAPH 2.0 OF BECHTEL TOPICAL REPORT BN-TOP-1 REVISION 1, NOVEMBER 1, 1972 "TESTING CRITERIA FOR INTEGRATED LEAKAGE RATE TESTING OF PRIMARY CONTAINMENT STRUCTURES FOR NUCLEAR POWER PLANTS." THE MASS POINT TECHNIQUE AS DESCRIBED IN ANSI/ANS-56.8-1981 PARAGRAPH 5.7.2 MAY BE USED IN LIEU OF THE TOTAL TIME OR POINT TO POINT METHODS.

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than either P_a, ~~[50 psig]~~, or at P_t, ~~[25 psig]~~, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

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SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a or 0.25 L_t;

2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a; or 0.75 L_t and 1.25 L_t.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, ~~[50 psig]~~ at intervals no greater than 24 months except for tests involving:

1) PERSONNEL &c. Air locks, and

2) Purge ~~supply~~ MAKEUP and exhaust isolation valves with resilient material seals,

~~3) Penetrations using continuous Leakage Monitoring Systems, and~~

~~4) Valves pressurized with fluid from a Seal System.~~

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.~~8.3 or 4.6.1.8.4~~ 7.2 as applicable; and

~~g. Type B periodic tests are not required for penetrations continuously monitored by the Containment Isolation Valve and Channel Weld Pressurization Systems provided the systems are OPERABLE by the requirements of Specification 4.6.1.4;~~

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SURVEILLANCE REQUIREMENTS (Continued)

~~h. Leakage from isolation valves that are sealed with fluid from a Seal System may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the Seal System and valves are pressurized to at least 1.10 P_a, [55 psig], and the seal system capacity is adequate to maintain system pressure for at least 30 days;~~

~~i. Type B tests for penetrations employing a continuous Leakage Monitoring System shall be conducted at P_a, [50 psig], at intervals no greater than once per 3 years; and~~

g ~~j~~. The provisions of Specification 4.0.2 are not applicable.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a. ~~[30] psig.~~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage is less than 0.01 L_a as determined by precision flow measurements when measured for at least 3.0 seconds with the volume between the seals at a constant pressure of ~~[50 psig]~~ 41 psig;

b. By conducting overall air lock leakage tests at not less than P_a, ~~[50 psig]~~, and verifying the overall air lock leakage rate is within its limit:

   1) At least once per 6 months,* and

   2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.**

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

*The provisions of Specification 4.0.2 are not applicable.

**This represents an exemption to Appendix J, paragraph III.D.2 of 10 CFR Part 50. ~~[Applicant must request this exemption.]~~

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CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.6.1.4 The Containment Isolation Valve and Channel Weld Pressurization Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Containment Isolation Valve or Channel Weld Pressurization System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The Containment Isolation Valve Pressurization System shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to 1.10 P_a, [55 psig], and has adequate capacity to maintain system pressure for at least 30 days.

4.6.1.4.2 The Containment Channel Weld Pressurization System shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to P_a, [50 psig], and has adequate capacity to maintain system pressure for at least 30 days.

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CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.~~8~~ 4 Primary containment internal pressure shall be maintained between 4.0 inch water guage and 1.9 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.~~8~~ 4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

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CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.~~8~~5 Primary containment average air temperature shall not exceed 120 °F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment average air temperature greater than 120 °F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.~~8~~5 The primary containment average air temperature shall be the arithmetical average of the temperatures OF THE OPERABLE MONITORS at the following location~~s~~ and shall be determined at least once per 24 hours:

Location

a. ELEVATION 327 FT. - 5 LOCATIONS

~~b.~~

~~c.~~

~~d.~~

~~e.~~

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CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY [Prestressed concrete containment with ungrouted tendons and typical dome.]

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With more than one tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specification 4.6.1.7, restore the containment vessel to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Containment Vessel Tendons. The containment vessel tendons' structural integrity shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a random but representative sample of at least 19 tendons (5 dome, 6 vertical, and 8 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and accceptable. Unless there is abnormal degradation of the containment vessel during the first three inspections, the sample population for subsequent inspections shall include at least 10 tendons (3 dome, 3 vertical, and 4 hoop);

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b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (dome, vertical, and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire or strand that:

1) The tendon wires or strands are free of corrosion, cracks, and damage,

2) There are not changes in the presence or physical appearance of the sheathing filler-grease, and

3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment vessel structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off stresses exceed the average minimum design value given below, which are adjusted to account for elastic losses; and

Dome	[143] ksi
Vertical	[147] ksi
Hoop	[140] ksi

e. Verifying the OPERABILITY of the sheathing filler grease by:

1) No voids in excess of 5% of the net duct volume i,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.7.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.7.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment vessel is at its maximum test pressure.

4.6.1.7.3 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

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CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY [Prestressed concrete containment with ungrouted tendons and hemispherical dome.]

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With more than one tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specification 4.6.1.7, restore the containment vessel to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Containment Vessel Tendons. The containment vessel tendons' structural integrity shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a random but representative sample of at least 11 tendons (4 inverted U and 7 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and accceptable. Unless there is abnormal degradation of the containment vessel during the first three inspections, the sample population for subsequent inspections shall include at least 6 tendons (3 inverted U and 3 hoop);

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SURVEILLANCE REQUIREMENTS (Continued)

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (inverted U and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire or strand that:

1) The tendon wires or strands are free of corrosion, cracks, and damage,

2) There are not changes in the presence or physical appearance of the sheathing filler-grease, and

3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment vessel structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off stresses exceed the average minimum design value given below, which are adjusted to account for elastic losses; and

Inverted U		[139] ksi
Hoop:	Cylinder	[147] ksi
	Dome	[134] ksi

e. Verifying the OPERABILITY of the sheathing filler grease by:

1) No voids in excess of 5% of the net duct volume i,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.7.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.7.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment vessel is at its maximum test pressure.

4.6.1.7.3 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

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CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY ~~[Reinforced concrete containment]~~

LIMITING CONDITION FOR OPERATION

3.6.1.~~7~~6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.~~7~~6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.~~7~~6.1. Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.~~7~~6.2 Reports. Any abnormal degradation of the containment vessel structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within ~~15~~ 30 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

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CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

MAKEUP

3.6.1.~~8~~7 Each containment purge ~~supply~~ and exhaust isolation valve shall be OPERABLE and:

PRE ENTRY MAKEUP

a. Each ~~[42-inch]~~ containment ~~shutdown~~ purge ~~supply~~ and exhaust isolation valve shall be closed and sealed closed. ~~and~~

~~b. The [8-inch]* containment purge supply and exhaust isolation valve(s) may be open for up to [1000]* hours during a calendar year provided no more than one pair (one supply and one exhaust) are open at one time.~~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

PRE ENTRY MAKEUP

a. With a ~~[42-inch]~~ containment purge ~~supply~~ and/or exhaust isolation valve open or not sealed closed, close and/or seal close that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~b. With the [8-inch]* containment purge supply and/or exhaust isolation valve(s) open for more than [1000]* hours during a calendar year, close the open [8-inch]* valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.~~

MAKEUP

b ~~c~~. With a containment purge ~~supply~~ and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7~~.8.3~~ ~~and/or 4.6.1.8.4~~, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

MAKEUP

4.6.1.~~8~~7.1 Each ~~[42-inch]~~ containment purge ~~supply~~ and exhaust isolation valve shall be verified to be sealed closed and closed at least once per 31 days.

~~4.6.1.8.2 The cumulative time that all (8-inch) purge supply and exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.~~

*For a 3-inch valve or less, the valves may be open continuously. For an 8-inch valve or less, the valves may be open for up to 1000 hours during a calendar year. For an 18-inch valve or less, the valves may be open for up to 500 hours during a calendar year. For a valve greater than 18 inches, the valve may be open for up to 250 hours during a calendar year. All valves that may be open during plant operations (MODE 1, 2, 3, or 4) must be qualified to close under postulated accident conditions. After operational experience, the licensee may request additional time for the qualified valves to be open during plant operations and shall provide justification for the requested additional time as indicated in the Bases.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.~~8~~7.~~3~~2 At least once per 6 months on a STAGGERED TEST BASIS, ~~the inboard and outboard isolation valves with resilient material seals in each sealed closed [42-inch] containment purge supply and exhaust penetration~~ EACH CONTAINMENT VENTILATION SYSTEM PENETRATION (WHICH IS ISOLATED BY 2-42" VALVES AND 2-8" VALVES) shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than ~~[0.05]~~ 0.06 L_a when pressurized to P_a.

~~4.6.1.8.4 At least once per 3 months each [8-inch] containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than [0.01] L_a when pressurized to P_a.~~

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CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM ~~[Credit taken for iodine removal]~~

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. REFER ALSO TO SPECIFICATION 3.6.2.3 ACTION.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. ~~By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to _____ psig when tested pursuant to Specification 4.0.5;~~

c. At least once per 18 months during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a _____ and Containment Spray Switchover to Containment Sump test signal, and CONTAINMENT SPRAY ACTUATION

2) Verifying that each spray pump starts automatically on a ______ test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

TESTING EACH CONTAINMENT SPRAY PUMP IN ACCORDANCE WITH SPECIFICATION 4.0.5.

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CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM [No credit taken for iodine removal]

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one Containment Spray System inoperable and at least [four] containment cooling fans OPERABLE, restore the inoperable Spray System to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two Containment Spray Systems inoperable and at least [four] containment cooling fans OPERABLE, restore at least one Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one Containment Spray System inoperable and one group of required containment cooling fans inoperable, restore either the inoperable Spray System or the inoperable group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both the inoperable Spray System and the inoperable group of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to ___ psig when tested pursuant to Specification 4.0.5;

c. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a _______ test signal, and

2) Verifying that each spray pump starts automatically on a ______ test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

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CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM ~~[OPTIONAL]~~

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 6000 and 6270 gallons of between 16 and 20 % by weight NaOH solution, and which is equivalent to between 85 and 88% indicated level

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a ______ test signal; and

CONTAINMENT SPRAY OR CONTAINMENT ISOLATION PHASE A, AS APPLICABLE

d. At least once per 5 years by verifying ~~each solution flow rate (to be determined during preoperational tests) from the following drain connections in the Spray Additive System:~~

~~1) [Drain line location] ______ ± ______ gpm, and~~

~~2) [Drain line location] ______ ± ______ gpm.~~

EACH EDUCTOR FLOW RATE IS GREATER THAN OR EQUAL TO 25gpm, USING THE RWST AS THE TEST SOURCE AND THROTTLED TO 17psig AT THE EDUCTOR INLET.

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CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM ~~[OPTIONAL] [Credit taken for iodine removal by spray systems]~~

LIMITING CONDITION FOR OPERATION

FOUR CONTAINMENT FAN COOLERS (AH-1, AH-2, AH-3 AND AH-4)
3.6.2.3 ~~[Two] independent groups of containment cooling fans~~ shall be OPERABLE with ~~[two] fan systems to each group. [Equivalent to 100% cooling capacity.]~~ ONE OF TWO FANS IN EACH COOLER CAPABLE OF OPERATION AT HALF SPEED. TRAIN SA CONSISTS OF AH-2 AND AH-3. TRAIN SB CONSISTS OF AH-1 AND AH-4.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one TRAIN ~~group~~ of the above required containment ~~cooling~~ fan COOLERS inoperable and both Containment Spray Systems OPERABLE, restore the inoperable group of FAN COOLERS ~~cooling fans~~ to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two TRAINS ~~groups~~ of the above required containment ~~cooling~~ fan COOLERS inoperable and both Containment Spray Systems OPERABLE, restore at least one TRAIN ~~group~~ of ~~cooling fans~~ FAN COOLERS to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required TRAINS OF FAN COOLERS ~~groups of cooling fans~~ to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one TRAIN ~~group~~ of the above required containment FAN COOLERS ~~cooling fans~~ inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable TRAIN ~~group~~ of containment FAN COOLERS ~~cooling fans~~ to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each TRAIN ~~group~~ of containment ~~cooling~~ fan COOLERS shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Starting each fan ~~group~~ from the control room, and verifying that each fan ~~group~~ operates for at least 15 minutes, and

2) Verifying a cooling water flow rate of greater than or equal to 1500 gpm to each cooler.

b. At least once per 18 months by verifying that each fan TRAIN ~~group~~ starts automatically on a SAFETY INJECTION test signal.

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CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM [OPTIONAL] [No credit taken for iodine removal by spray systems]

LIMITING CONDITION FOR OPERATION

3.6.2.3 [Two] independent groups of containment cooling fans shall be OPERABLE with [two] fan systems to each group. [Equivalent to 100% cooling capacity.]

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one group of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore the inoperable group of cooling fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore at least one group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required containment cooling fans inoperable and one Containment Spray System inoperable, restore either the inoperable group of containment cooling fans or the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both the inoperable group of containment cooling fans and the inoperable Spray System to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Starting each fan group from the control room and verifying that each fan group operates for at least 15 minutes, and

2) Verifying a cooling water flow rate of greater than or equal to ____ gpm to each cooler.

b. At least once per 18 months by verifying that each fan group starts automatically on a ______ test signal.

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CONTAINMENT SYSTEMS

3/4.6.3 IODINE CLEANUP SYSTEM [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.6.3 Two independent Iodine Cleanup Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Iodine Cleanup System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 Each Iodine Cleanup System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedures guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ________ cfm ± 10%.

2) Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and

3) Verifying a system flow rate of _____ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than [6] inches Water Gauge while operating the system at a flow rate of ______ cfm ± 10%;

2) Verifying that the system starts on either a Safety Injection test signal or on a Containment Pressure-High test signal;

3) Verifying that the filter cooling bypass valves can be opened by operator action; and

4) Verifying that the heaters dissipate ______ ± ______ kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and by leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of ______ cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of _____ cfm ± 10%.

*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).

**Value applicable will be determined by the following equation:

$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

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3/4.6.~~4~~ 3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.~~4~~ 3 The containment isolation valves specified ~~in Table 3.6-1~~ in FSAR Table 6.2.4-1* shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment isolation valve(s) specified in ~~Table 3.6-1~~ FSAR Table 6.2.4-1* inoperable, maintain at least one isolation valve OPERABLE in each affected penetration ** that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. THE PROVISIONS OF SPECIFICATION 3.0.4 ARE NOT APPLICABLE

SURVEILLANCE REQUIREMENTS

4.6.~~4~~ 3.1 ~~The~~ THE POWER OPERATED OR AUTOMATIC (WITH THE EXCEPTION OF SELF ACTUATING VALVES) CONTAINMENT isolation valves specified in ~~Table 3.6-1~~ FSAR TABLE 6.2.4-1* shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

* EXCEPT FOR MSIV'S WHICH ARE COVERED BY SPECIFICATION 3.7.1.5.

** LINES WITH SINGLE ISOLATION VALVES MEET THESE ACTIONS BY CLOSING AND DEACTIVATING THE ISOLATION VALVE AS IN ACTION b. ONLY.

SURVEILLANCE REQUIREMENTS (Continued)

FSAR TABLE 6.2.4-1

4.6.~~4~~3.2 Each isolation valve specified in ~~Table 3.6-1~~ shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Containment Isolation Phase "A" ~~Isolation~~ test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Containment Isolation Phase "B" ~~Isolation~~ test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation ~~Purge and Exhaust~~ Isolation test signal, ~~each purge and exhaust~~ valve actuates to its isolation position.

4.6.~~4~~3.3 The isolation time of each power-operated or automatic valve of ~~Table 3.6-1~~ FSAR TABLE 6.2.4-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

POWER OPERATED OR AUTOMATIC (WITH THE EXCEPTION OF SELF ACTUATING VALVES) CONTAINMENT

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation		
a.		
b.		
2. Phase "B" Isolation		
a.		
b.		
3. Containment Purge and Exhaust		
a.		
b.		
4. Manual		
a.		
b.		
5. Other		
a.		
b.		

* May be opened on an intermittent basis under administrative control.
** Not subject to Type C leakage tests.
***The provisions of Specification 3.0.4 are not applicable.

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CONTAINMENT SYSTEMS

3/4.6.~~5~~4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.~~5~~4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.~~5~~4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ~~ANALOG~~ CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing:

a. ~~One~~ Two volume percent hydrogen, balance nitrogen, and

b. ~~Four~~ Six volume percent hydrogen, balance nitrogen.

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CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

4

3.6.~~5~~.2 Two independent Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Recombiner System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4

4.6.~~5~~.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

a. At least once per 6 months ~~by~~ verifying, during a Hydrogen Recombiner System functional test, that the minimum heater sheath temperature increases to greater than or equal to [700]°F within [90] minutes. Upon reaching [700]°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to [60] kW, and

b. At least once per 18 months by:

   1) Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

   2) Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

   3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

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CONTAINMENT SYSTEMS

HYDROGEN PURGE CLEANUP SYSTEM [If less than 2 hydrogen recombiners available]

LIMITING CONDITION FOR OPERATION

3.6.5.3 A Hydrogen Purge Cleanup System shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the Hydrogen Purge Cleanup System inoperable, restore the Hydrogen Purge Cleanup System to OPERABLE status within 30 days or be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3 The Hydrogen Purge Cleanup System shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance of the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ______ cfm ± 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%: and

   3) Verifying a system flow rate of _____ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than [6] inches Water Gauge while operating the system at a flow rate of ___ cfm ± 10%,

2) Verifying that the filter cooling bypass valves can be manually opened, and

3) Verifying that the heaters dissipate _____ ± _____ kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of _____ cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of _______ cfm ± 10%.

---

*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)

**Value applicable will be determined by the following equation:

$P = \frac{100\% - E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

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CONTAINMENT SYSTEMS

HYDROGEN MIXING SYSTEM [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent Hydrogen Mixing Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Mixing System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each Hydrogen Mixing System shall be demonstrated OPERABLE:

a. At least once per 92 days on a STAGGERED TEST BASIS by starting each system from the control room and verifying that the system operates for at least 15 minutes, and

b. At least once per 18 months by verifying a system flow rate of at least ________ cfm.

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3/4.6.6 PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.6.6 Two independent Penetration Room Exhaust Air Cleanup Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Penetration Room Exhaust Air Cleanup System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6 Each Penetration Room Exhaust Air Cleanup System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ______ cfm ± 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.5.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a system flow rate of ___ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than [6] inches Water Gauge while operating the system at a flow rate of _____ cfm ± 10%,

2) Verifying that the system starts on a Safety Injection test signal,

3) Verifying that the filter cooling bypass valves can be manually opened, and

4) Verifying that the heaters dissipate _____ ± ______ kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of _____ cfm ± 10%, and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of _______ cfm ± 10%.

---

*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)

**Value applicable will be determined by the following equation:

$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

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3/4 6.~~X~~5 VACUUM RELIEF SYSTEM ~~VALVES [OPTIONAL]~~

LIMITING CONDITION FOR OPERATION

3.6.~~X~~5 The ~~primary~~ containment ~~to atmosphere~~ vacuum relief SYSTEM ~~valves~~ shall be OPERABLE with an Actuation Setpoint of GREATER ~~less~~ than or equal to ~~___ psid.~~ −2.5 INCHES WATER GAUGE DIFFERENTIAL PRESSURE (CONTAINMENT PRESSURE LESS ATMOSPHERIC PRESSURE)

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one ~~primary~~ containment ~~to atmosphere~~ vacuum relief SYSTEM ~~valve~~ inoperable, restore the SYSTEM ~~valve~~ to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.~~X~~5 No additional requirements other than those required by Specification 4.0.5.

Capitalize the Titles of the Following Systems and component names in the Technical Specifications

Main Steam Line Code Safety Valves

Steam Generator

Auxiliary Feedwater System (Pumps)

Condensate Storage Tank

Emergency Service Water System

Main Steam Line Isolation Valve

Component Cooling Water System

Safety Injection

Ultimate Heat Sink

Auxiliary Reservoir

Main Reservoir

Control Room Emergency Filtration System

Reactor Auxiliary Building Emergency

Exhaust System

Fire Protection Water Supply and Distribution System

Essential Services Chilled Water System

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3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator ~~of an unisolated reactor coolant loop~~ shall be OPERABLE with lift settings as specified in Table 3.7-~~3~~2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. ~~With (n) reactor coolant loops and associated steam generators in operation and~~ With one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~b. With (n-1) reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves associated with an operating loop inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

b ~~c~~. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5.

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TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING ~~N~~ 3 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	~~[87]~~
2	~~[64]~~
3	~~[42]~~

TABLE 3.7-2

~~MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING N-1 LOOP OPERATION~~

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR*	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
~~1~~	~~[52]~~
~~2~~	~~[38]~~
~~3~~	~~[25]~~

~~*At least two safety valves shall be OPERABLE on the non-operating steam generator.~~

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TABLE 3.7-~~3~~2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER				LIFT SETTING (± 1%)*	ORIFICE SIZE (IN.²)
STEAM GENERATOR	A	B	C		
~~1.~~	1MS-43	1MS-44	1MS-45	1170 psig	16.0
~~2.~~	1MS-46	1MS-47	1MS-48	1185 psig	16.0
~~3.~~	1MS-49	1MS-50	1MS-51	1200 psig	16.0
~~4.~~	1MS-52	1MS-53	1MS-54	1215 psig	16.0
	1MS-55	1MS-56	1MS-57	1230 psig	16.0

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY witin 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1510 psig at a recirculation flow of greater than or equal to 50 gpm;

2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1455 psig ~~at~~ ON A RECIRCULATION flow of greater than or equal to 100 gpm when the secondary steam supply pressure is greater than ____ psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

4) Verifying that each automatic valve in the flow path (EXCEPT PRESSURE CONTROL VALVES) is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER.

b. At least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal.

4.7.1.2.2~~*~~ An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying normal flow to ~~each~~ AT LEAST TWO steam generatorS FROM AT LEAST ONE AUXILARY FEEDWATER PUMP.

~~*This is applicable only for plants that do not use auxiliary feedwater for STARTUP/SHUTDOWN operations.~~

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CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained water volume of at least 254,000 gallons of water WHICH IS EQUIVALENT TO 60% INDICATED LEVEL.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the CST inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the ~~[alternate water source]~~ EMERGENCY SERVICE WATER SYSTEM as a backup supply to the auxiliary feedwater pumps and restore the CST to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The CST shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 ~~The [alternate water source]~~ THE EMERGENCY SERVICE WATER SYSTEM shall be demonstrated OPERABLE at least once per 12 hours by ~~[method dependent upon alternate source]~~ VERIFYING THAT EACH VALVE THAT IS REQUIRED TO PERMIT THE EMERGENCY SERVICE WATER SYSTEM TO SUPPLY WATER TO THE AUXILARY FEEDWATER PUMPS IS OPEN. whenever the ~~[alternate water source]~~ EMERGENCY SERVICE WATER SYSTEM is the supply source for the auxiliary feedwater pumps.

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SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.1 microCurie/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.1 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

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TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination* or ~~2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration~~	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the secondary coolant except for radionuclides with half-lives less than 10 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level.

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MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than [70]°F when the pressure of either coolant in the steam generator is greater than [200] psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to [200] psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than [200] psig at least once per hour when the temperature of either the reactor or secondary coolant is less than [70]°F.

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PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two ~~independent~~ component cooling water PUMPS#, HEAT EXCHANGERS AND ESSENTIAL FLOW PATHS ~~loops~~ shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position~~; and~~;

c ~~b~~. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment or isolating non safety-related components actuates to its correct position on a Safety Injection test signal, and

2) Each Component Cooling Water System pump starts automatically on a Safety Injection test signal.

b. At least once per 31 days by performing an OPERATIONAL TEST of the surge tank level indication which provides automatic isolation of cooling water to the Gross Failed Fuel Detector; and

3) Each automatic valve serving the Gross Failed Fuel Detector actuates to its correct position on a Low Surge Tank Level Test signal

\# The breaker for CCW Pump 1C-SAB shall not be racked in to either power source (SA or SB) unless the breaker from the applicable CCW pump (1A-SA or 1B-SB) is racked out.

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3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent EMERGENCY Service Water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one EMERGENCY Service Water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two EMERGENCY Service Water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment OR ISOLATING NON SAFETY PORTIONS OF THE SYSTEM actuates to its correct position on a SAFETY INJECTION test signal, and

2) Each Service Water System pump starts automatically on a SAFETY INJECTION test signal.

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3/4.7.5 ULTIMATE HEAT SINK [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum Auxilary Reservoir water level at or above elevation 250 feet Mean Sea Level, USGS datum, and a Minimum Main Reservoir water level at or above 205.7 feet Mean Sea Level USGS datum, and

b. An ~~average~~ water temperature of Respective Auxilary Reservoir as measured at the Intake Structure less than or equal to 95 °F.

c. An ~~average~~ water temperature of the Auxilary Reservoir as measured at the intake structure greater than 35 °F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the requirements of ~~the above~~ specifications 3.7.5.a or 3.7.5.b not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the requirements of specification 3.7.5.c not satisfied, start at least one Emergency Service Water Pump taking suction from the Auxilary Reservoir and discharging to the Auxilary Reservoir.

c. The requirements of 3.0.4 and 4.0.4 do not apply if Action b is in effect.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

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3/4.7.6 FLOOD PROTECTION [OPTIONAL*]

LIMITING CONDITION FOR OPERATION

3.7.6 Flood protection shall be provided for all Safety-Related Systems, components, and structures when the water level of the ______ [usually the ultimate heat sink] exceeds ______ Mean Sea Level, USGS datum, at ________.

APPLICABILITY: At all times.

ACTION:

With the water level at ______ above elevation ______ Mean Sea Level, USGS datum:

a. [Be in at least HOT STANDBY within 6 hours and in at least COLD SHUTDOWN within the following 30 hours], and

b. Initiate and complete within ______ hours, the following flood protection measures:

   1. [Plant dependent], and

   2. [Plant dependent].

SURVEILLANCE REQUIREMENTS

4.7.6 The water level at _____ shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation _____ Mean Sea Level, USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation _____ Mean Sea Level, USGS datum.

* This specification not required if the facility design has adequate passive flood control protection features sufficient to accommodate the Design Basis Flood identified in Regulatory Guide 1.59, August 1973.

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3/4.7.~~X~~6 CONTROL ROOM EMERGENCY FILTRATION ~~AIR CLEANUP~~ SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~X~~6 Two independent Control Room Emergency FILTRATION ~~Air Cleanup~~ Systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room Emergency FILTRATION ~~Air Cleanup~~ System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one Control Room Emergency FILTRATION ~~Air Cleanup~~ System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency FILTRATION ~~Air Cleanup~~ System in the recirculation mode.

b. With both Control Room Emergency FILTRATION ~~Air Cleanup~~ Systems inoperable, or with the OPERABLE Control Room Emergency FILTRATION ~~Air Cleanup~~ System, required to be in the recirculation mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.~~X~~6 Each Control Room Emergency FILTRATION ~~Air Cleanup~~ System shall be demonstrated OPERABLE:

~~a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to [80]°F;~~

a ~~b~~. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

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SURVEILLANCE REQUIREMENTS (Continued)

b ~~f~~. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[**]%~~ .05% and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revisions 2, March 1978, and the system flow rate is 4000 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~[**]%~~ .2%; and

3) Verifying a system flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c ~~d~~. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~[**]%~~ .2%;

d ~~e~~. At least once per 18 months by:

1) Verifying that the TOTAL pressure drop across the ~~combined HEPA filters and charcoal adsorber banks is less than [6]~~ CONTROL ROOM EMERGENCY FILTRATION UNIT IS LESS THAN 8.49 inches Water Gauge while operating the system at a flow rate of 4000 cfm ± 10%;

2) Verifying that on a ~~Containment Phase "A" Isolation and High Smoke Density~~ SAFETY INJECTION AND HIGH RADIATION test signal, the system automatically switches into ~~a~~ AN ISOLATION WITH recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks;

3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to ~~[1/8]~~ inch Water Gauge at less than or equal to a pressurization flow of 400 cfm relative to adjacent areas during system operation;

4) Verifying that the heaters dissipate 14 ± 1.4 kW when tested in accordance with ANSI N510-1975; and

5) Verifying that on a High Chlorine ~~Toxic Gas~~ test signal, the system automatically switches into ~~a~~ AN ISOLATION WITH recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks within ~~[15]~~ seconds.

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SURVEILLANCE REQUIREMENTS (Continued)

e ~~f.~~ After each complete or partial UNIT replacement of a HEPA filter bank, by verifying that the ~~cleanup system~~ satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.5% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 4000 cfm ± 10%; and

f ~~g.~~ After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

~~*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)~~

~~**Value applicable will be determined by the following equation:~~

~~$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).~~

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3/4.7.~~8~~7 ~~ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM~~ REACTOR AUXILARY BUILDING (RAB) EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~8~~7 Two independent ~~ECCS Pump Room Exhaust Air Cleanup~~ RAB Emergency Exhaust Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one ~~ECCS Pump Room Exhaust Air Cleanup~~ RAB Emergency Exhaust System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.~~8~~7 Each ~~ECCS Pump Room Exhaust Air Cleanup~~ RAB Emergency Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the ~~system~~ unit flow rate is 6800 acfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~[**]%~~ 0.2%; and

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a ~~system~~ unit flow rate of 6800 acfm + 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~[**]%~~; 0.2%

d. At least once per 18 months by:

1) Verifying that the total pressure drop across the ~~combined HEPA filters and charcoal adsorber banks~~ RAB Emergency Exhaust Unit is less than ~~[6]~~ 9.22 inches Water Gauge while operating the ~~system~~ unit at a flow rate of 6800 acfm ± 10%,

2) Verifying that the system starts on a Safety Injection test signal,

3) Verifying that the system maintains the ECCS pump room at a negative pressure of greater than or equal to ~~[~~1/8~~]~~ inch Water Gauge relative to the outside atmosphere,

4) Verifying that the filter cooling bypass valve~~s~~ ~~can be manually opened~~ is locked opened, and

5) Verifying that the heaters dissipate 40 ± 4 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial unit replacement of a HEPA filter bank, by verifying that the ~~cleanup system~~ satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the ~~system~~ unit at a flow rate of 6800 acfm ± 10%; and

f. After each complete or partial unit replacement of a charcoal adsorber bank, by verifying that the ~~cleanup system~~ satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the ~~system~~ unit at a flow rate of 6800 acfm ± 10%.

~~*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)~~

~~**Value applicable will be determined by the following equation:~~

~~$P = \frac{100\% - E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).~~

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PLANT SYSTEMS

3/4.7.~~9~~ 8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.~~9~~ 8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure of failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.~~9~~ 8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in lieu of the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type [on any system] are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection [of that system] shall be performed at the first refueling outage. Otherwise, subsequent visual inspections [of a given system] shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type [on Any System] per Inspection Period	Subsequent Visual Inspection Period* **
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

*The inspection interval for each type of snubber [on a given system] shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found [on that system].

**The provisions of Specification 4.0.2 are not applicable.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. 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 for other snubbers irrespective of type [on that system] that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.~~8~~f. All 8 snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. [For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems.]

d. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers of each type shall be tested using one of the following sample plans. The sample plan for each type shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.~~8~~f., an additional 10% 8 of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

e. Functional Tests (Continued)

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.~~9~~f. 8 The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

SURVEILLANCE REQUIREMENTS (Continued)

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.~~6~~e. for snubbers not meeting the functional test acceptance criteria. 8

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SURVEILLANCE REQUIREMENTS (Continued)

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.~~3~~. 2

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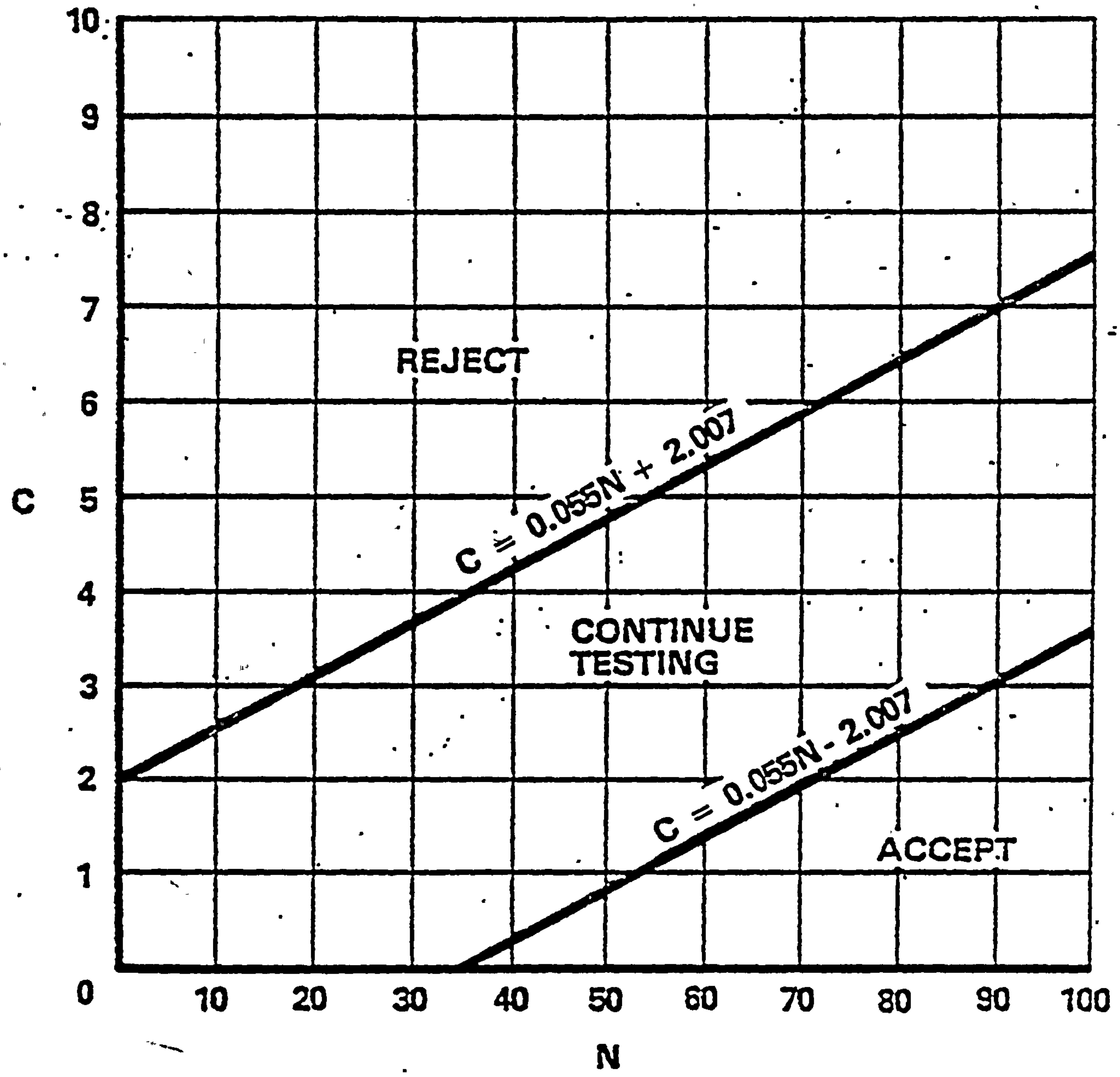

FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

3/4.7.~~10~~ 9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.~~10~~ 9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or ~~5~~ 10 microCuries of alpha emitting Byproduct material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

1. Decontaminate and repair the sealed source, or

2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.~~10~~ 9.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.~~10~~ 9.2 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.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

1) With a half-life greater than 30 days (excluding Hydrogen 3), and

2) In any form other than gas.

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SURVEILLANCE REQUIREMENTS (Continued)

b. 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 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. 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.

4.7.~~10~~ 9.3 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 greater than or equal to 0.005 microCurie of removable contamination.

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3/4.7.~~11~~ 10 FIRE SUPPRESSION SYSTEMS

FIRE ~~SUPPRESSION WATER SYSTEM~~ PROTECTION WATER SUPPLY AND DISTRIBUTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~11~~ 10.1 The Fire ~~Suppression Water System~~ Protection Water Supply and Distribution System shall be OPERABLE with:

a. At least [two] fire ~~suppression~~ pumps, each with a capacity of 2100 ~~[2500]~~ gpm, with their discharge aligned to the fire suppression header,

b. ~~Separate water supplies, each with a minimum contained volume of ____ gallons, and~~ The Auxilary Reservoir water level shall be maintained in accordance with Specification 3.7.5, and

c. An OPERABLE flow path capable of taking suction from the Auxilary Reservoir ~~tank and the ____ tank~~ and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each ~~Deluge or~~ Spray System required to be OPERABLE per Specifications 3.7.~~11~~ 10.2, 3.7.~~11.5,~~ 10.3 and 3.7.~~11.6.~~ 10.4

APPLICABILITY: At all times.

ACTION:

a. With one pump ~~and/or one water supply~~ inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire ~~Suppression~~ Protection Water Supply and Distribution System otherwise inoperable, establish a backup ~~Fire Suppression Water~~ System within 24 hours.

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SURVEILLANCE REQUIREMENTS

4.7.~~11~~ 10.1.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

~~a. At least once per 7 days by verifying the contained water supply volume,~~

a ~~b~~. At least once per 31 days ~~on a STAGGERED TEST BASIS~~ by starting each electric motor-driven pump and operating it for at least 15 minutes on ~~recirculation~~ RELIEF VALVE flow,

b ~~c~~. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

~~d. [At least once per 6 months by performance of a system flush,]~~

c ~~d~~ ~~e~~. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

d ~~e~~ ~~f~~. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

~~1) Verifying that each automatic valve in the flow path actuates to its correct position,~~

1 ~~2~~) Verifying that each pump develops at least ~~[2500]~~ 2100 gpm at a discharge ~~system head~~ pressure of ~~[250] feet,~~ 131 psig

2 ~~3~~) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3 ~~4~~) Verifying that each fire suppression pump starts ~~[sequentially]~~ to maintain the Fire Suppression Water System pressure greater than or equal to 80 psig.

e ~~g~~. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

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SURVEILLANCE REQUIREMENTS (Continued)

10
4.7.~~11~~.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying: - 130

1) The fuel storage tank contains at least __ gallons of fuel, and

2) The diesel starts from ambient conditions and operates for at least 30 minutes on ~~recirculation~~ RELIEF VALVE flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-0270-1975 is within the acceptable limits specified in Table 1 of ASTM 0975-1977 when checked for viscosity and water and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

10
4.7.~~11~~.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The electrolyte level of each battery is above the plates, and

2) The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery, and

c. At least once per 18 months by verifying that:

1) The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

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~~SPRAY AND/OR SPRINKLER SYSTEMS~~ PREACTION AND MULTICYCLE SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.~~11~~10.2 The following ~~Spray and/or~~ PRE-ACTION AND MULTICYCLE Sprinkler Systems LISTED ON TABLE 3.7-3 shall be OPERABLE:

~~a. [Plant dependent - to be listed by name and location.]~~

~~b.~~

~~c.~~

APPLICABILITY: Whenever equipment protected by ~~the Spray/Sprinkler~~ A PRE-ACTION OR MULTICYCLE SPRINKLER System is required to be OPERABLE.

ACTION:

a. With one or more of the above required ~~Spray and/or~~ PRE-ACTION OR MULTICYCLE Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.~~11~~10.2 Each of the above required ~~Spray and/or~~ PREACTION OR MULTICYCLE Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

1) By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a THERMAL test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

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TABLE 3.7-3

~~INSERT A - PLANT SYSTEMS~~

PRE-ACTION AND MULTICYCLE SPRINKLER SYSTEMS

		LOCATION/ELEVATION
a.	Airborne Radioactivity Removal Unit - 1A Sprinkler (1-C-1-CHFA)	C.B. /221
b.	Airborne Radioactivity Removal Unit-1B Sprinkler (1-C-1-CHFB)	C.B. /221
c.	Electrical Cable Penetration Area-1A Sprinkler (1-C-~~4~~ 1-EPA)	C.B. /261
d.	Electrical Cable Penetration Area-1B Sprinkler (1-C-~~3~~ 1-EPB)	C.B. /261
e.	Containment Spray and RHR Pump Room 1A Sprinkler (1-A-1-PA)	RAB /190
f.	Containment Spray and RHR Pump Room 1B Sprinkler (1-A-1-PB)	RAB /190
g.	AUX. Feed Water Pumps and Component Cooling Water Heat Exchanger and Pumps SPrinkler (1-A-3-PB)	RAB /236
h.	Decontamination Area and Corridor Cable Tray Sprinkler (1-A-3-~~CMB~~) COMB	RAB /236
i.	Letdown Heat Exchanger Area, Corridor Cable Tray Sprinkler (1-A-3-~~COM~~) COME	RAB /236
j.	Recycle Holdup Tank Area, Corridor Cable Tray Sprinkler (1-A-3-COM1)	RAB /236
k.	HVAC Chiller Equipment Area and Cable Tray Sprinkler (1-A-4-CHLR)	RAB /261
l.	Boric Acid Equipment Area, Corridor Cable Tray Sprinkler (1-A-4-COMB)	RAB /261
m.	Corridor Cable Tray Sprinkler (1-A-4-COME)	RAB /261 Column 43, E to H
n.	Corridor Cable Tray Sprinkler (1-A-4-COMI)	RAB /261 Column 43, I to L
o.	Charcoal Filter Room 1A Sprinkler (1-A-4-CHFA)	RAB /261

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TABLE 3.7-3
~~INSERT A — PLANT SYSTEMS~~ (Cont'd)

		LOCATION/ELEVATION
p.	Charcoal Filter Room 1B Sprinkler (1-A-4-CHFB)	RAB /261
q.	Electrical Penetration Area SA Sprinkler (1-A-EPA)	RAB /261
r.	Electrical Penetration Area SB Sprinkler (1-A-EPB)	RAB /261
s.	Cable Spreading Room A Sprinkler (1-A-CSRA)	RAB /286
t.	Cable Spreading Room B Sprinkler (1-A-CSRB)	RAB /286
u.	HVAC Equipment Room Sprinkler (12-A-6-HV7)	RAB /305
v.	Emergency Exhaust System E-12 (5-F-3-CHFA)	FHB /261
x.	Emergency Exhaust System E-13 (5-F-3-CHFB)	FHB /261
y.	Fuel Pool Cooling Heat Exchangers and Pumps (5-F-2-FPC)	FHB /236
z.	Diesel Generator 1A-Sprinkler (1-D-1-DGA-RM)	DGB /261
aa.	Diesel Generator 1B-Sprinkler (1-D-1-DGB-RM)	DGB /261
bb.	Diesel Fuel Oil Day Tank 1A-Sprinkled (1-D-1-DGA-TK)	DGB /280
cc.	Diesel Fuel Oil Day Tank 1B-Sprinkler (1-D-1-DGB-TK)	DGB /280
dd.	Diesel Oil Pump Room 1A-Sprinkler (1-O-PA)	Diesel Fuel Oil Storage Tank Area /242.25
ee.	Diesel Oil Pump Room 1B-Sprinkler (1-O-PB)	Diesel Fuel Oil Storage Tank Area /242.25

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SURVEILLANCE REQUIREMENTS (Continued)

2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity; and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air or water flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

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CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following High Pressure and Low Pressure CO_2 Systems shall be OPERABLE:

a. [Plant dependent - to be listed by name and location.]

b.

c.

APPLICABILITY: Whenever equipment protected by the CO_2 Systems is required to be OPERABLE.

ACTION:

a.. With one or more of the above required CO_2 Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.3.1 Each of the above required CO_2 Systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

4.7.11.3.2 Each of the above required Low Pressure CO_2 Systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than ____ and pressure to be greater than ____ psig, and

b. At least once per 18 months by verifying:

1) The system, including valves, associated ventilation system fire dampers, and fire door release mechanisms, actuates manually and automatically upon receipt of a simulated actuation signal, and

2) Flow from each nozzle during a "Puff Test."

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SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.3.3 Each of the above required High Pressure CO_2 Systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying the CO_2 storage tank weight to be at least 90% of full charge weight, and

b. At least once per 18 months by:

1) Verifying the system, including associated ventilation system fire dampers and fire door release mechanisms, actuates manually and automatically upon receipt of a simulated actuation signal; and

2) Performance of a flow test through headers and nozzles to assure no blockage.

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The following Halon Systems shall be OPERABLE:

a. [Plant dependent - to be listed by name and location.]

b.

c.

APPLICABILITY: Whenever equipment protected by the Halon System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the above required Halon Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 6 months by verifying Halon storage tank weight to be at least 95% of full charge weight [or level] and pressure to be at least 90% of full charge pressure, and

c. At least once per 18 months by:

1) Verifying the system, including associated Ventilation System fire dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

2) Performance of a flow test through headers and nozzles to assure no blockage.

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PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.~~11.5~~ 10.3 The fire hose stations given in Table 3.7-4 shall be OPERABLE.*

APPLICABILITY: Whenever SAFE SHUTDOWN equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations given in Table 3.7-4 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above ACTION requirement shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.~~11.5~~ 10.3 Each of the fire hose stations given in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station.

b. At least once per 18 months, by:

1) Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

2) Removing the hose for inspection and re-racking, and

3) Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years, by:

1) Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

2) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

* Fire hose stations within the Containment are required to be OPERABLE only during refueling and maintenance outages.

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REFER TO NEXT PAGES
FOR TABLE 3.7-4 DATA.

TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION*	ELEVATION	HOSE RACK NUMBER

*List all Fire Hose Stations required to ensure the OPERABILITY of safety-related equipment.

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~~Insert B Plant Systems~~

Table 3.7-~~3~~4

Fire Hose Stations

LOCATION[1]	ELEVATION	HOSE RACK NO.
CB	221	221-C-4
CB	221	221-C-12
CB	221	221-C-19
CB	236	236-C-4
CB	236	236-C-12
CB	236	236-C-19
CB	261	261-C-4
CB	261	261-C-12
CB	261	261-C-19
CB	286	286-C-4
CB	286	286-C-12
CB	286	286-C-19
RAB	190	190-G-16
RAB	190	190-G-38
RAB	216	216-G-16
RAB	216	216-Fz-27
RAB	216	216-G-38
RAB	216	216-Gy-13
RAB	236	236-Gy-13
RAB	236	236-G-16
RAB	236	236-Fz-27
RAB	236	236-D-27
RAB	236	236-G-38
RAB	236	236-Kz-31
RAB	236	236-C-39
RAB	236	236-Fw-43
RAB	236	236-Jz-43
RAB	236	236-E-15
RAB	261	261-Gy-13
RAB	261	261-E-15
RAB	261	261-G-16
RAB	261	261-D-27
RAB	261	261-Kz-31
RAB	261	261-G-38
RAB	261	261-C-39
RAB	261	261-Fw-42
RAB	286	286-C-15
RAB	286	286-E-15
RAB	286	286-~~K~~G-16
RAB	286	286-E-38
RAB	286	286-C-39
RAB	286	286-~~Jy-41~~ Jv-41
RAB	286	286-Fw-42
~~RAB~~	~~286~~	~~286-Fw-44~~
~~RAB~~	~~286~~	~~286-Jy-45~~

[1]CB - Containment Building FHB - Fuel Handling Building
RAB - Reactor Auxiliary Building DGB - Diesel Generator Building

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~~Insert B Plant Systems (Cont'd)~~

Table 3.7-34 (cont'd)

Fire Hose Stations

LOCATION[1]	ELEVATION	HOSE RACK NO.
RAB	261	261-Jz-43
RAB	261	261-Fw-43
RAB	305	305-C-39
RAB	305	305-I-41
RAB	305	305-Fw-43
FHB	236	236-L-41
FHB	236	236-L-45
FHB	261	261 ~~236~~-L-41
FHB	261	261 ~~236~~-L-45
FHB	286	286-L-27
FHB	286	286-N-36
FHB	286	286-L-43
FHB	286	286-N-51
FHB	286	286-L-65
FHB	286	286-N-71
FHB	286	286-L-75y.
DGB	261	261-C-2
DGB	261	261-C-4
DGB	261	261-B-1
DGB	261	261-B-2

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[1]CB - Containment Building FHB - Fuel Handling Building
RAB - Reactor Auxiliary Building DGB - Diesel Generator Building

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YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.~~11.6~~ 10.4 The yard fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses given in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.~~11.6~~ 10.4 Each of the yard fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days, by visual inspection of the hydrant hose house to assure all required equipment is at the hose house,

~~b. At least once per 6~~ year ~~months (once during March, April, or May and once during September, October, or November), by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged, and~~

b ~~c~~. At least once per 12 months by:

1) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater,

2) Inspecting all the gaskets and replacing any degraded gaskets in the couplings, and

3) Performing a flow check of each hydrant to verify its OPERABILITY.

4) Visually inspecting each yard hydrant and verifying that the hydrant is dry and is not damaged (to be performed during September, October or November).

5) Visually inspecting each yard hydrant and verifying that it is not damaged (to be performed during March, April or May).

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TABLE 3.7-5

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION*	HYDRANT NUMBER

REFER TO NEXT PAGE
FOR TABLE DATA

*List all Yard Fire Hydrants and Hydrant Hose Houses required to ensure the OPERABILITY of safety-related equipment:

~~Insert C-Plant Systems~~

Table 3.7-~~4~~5

Yard Fire Hydrant and Associated

LOCATION		HYDRANT NO.
Emergency Service Water Intake Structure		1-4AJ-NNS
Emergency Service Water Screening Structure		1-4AI-NNS
Diesel Generator Building	North Side	1-4B-NNS
	South Side	1-4A-NNS
Diesel Fuel Oil Storage Tank Building	East Side	1-4V-NNS
	West Side	1-4H-NNS

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PLANT SYSTEMS

3/4.7.~~12~~ 11 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

REQUIRED FOR IN THE EVENT OF FIRE

3.7.~~12~~ 11 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems ~~important to~~ safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping, and ventilation duct penetration seals shall be OPERABLE.

APPLICABILITY: ~~At all times.~~ WHENEVER THE EQUIPMENT IN AN AFFECTED AREA IS REQUIRED TO BE OPERABLE.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.~~12~~ 11.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper and associated hardware, and

c. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration will be inspected every 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.~~12~~11.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days,

b. That each locked closed fire door is closed at least once per 7 days,

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours, and a functional test is performed at least once per 18 months, and

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

3/4.7.~~13~~ 12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.~~13~~ 12 The temperature of each area shown in Table 3.7-6 shall not be exceeded for more than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by more than 30°F, prepare and submit a Special Report as required by ACTION a. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.~~13~~ 12 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limit at least once per 12 hours.

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TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
1.	
2.	
3.	
4.	
5.	

REFER TO NEXT PAGE
FOR TABLE DATA

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~~Insert G Plant Systems~~
Table 3.7-~~5~~6

Area	Maximum Temperature Limit (°F)
REACTOR AUXILIARY BUILDING	
1. Control Room Envelope, (El. 305')	(later)
2. Process I&C, Room (El. 305')	(later)
3. Rod Control Cabinets Area (El. 305')	104
4. A&B Battery Rooms (El. 286')	(later)
5. A&B Switchgear Rooms (El. 286')	(later)
6. Main Steam, Feedwater Pipe Tunnel (El. 286' & 261')	111
7. SA&SB Electrical Penetration Areas (El. 261')	104
8. Area with MCC 1A35MSA and 1B35SB	104
9. HVAC Chillers, Auxiliary FW Piping & Valve Area (El. 261')	104
10. CCW Pumps, CCW Hx, Auxiliary FW Pumps Area (El. 236')	104
11. 1A-SA, 1B-SB, 1C-SAB and Spare Charging Pump Rooms (El. 236')	104
12. Service Water Booster Pump 1B-SB	104
13. Mechanical and Electrical Penetration Areas (El. 236')	104
14. Containment Spray Additive Tank, and H&V Equipment Area (El. 216')	104
15. Trains A&B Containment Spray Pump, RHR Pump, H&V Equipment Areas	104
FUEL HANDLING BUILDING	
16. Trains A&B Emergency Exhaust System Areas (El. 261')	104
17. Fuel Pool Cooling Pump and Heat Exchanger Area (El. 236')	104
WASTE PROCESSING BUILDING	
18. H&V Equipment Rooms (El. 236')	104
MISCELLANEOUS	
19. Condensate Storage Tank Area (El. 261')	104
20. Diesel Fuel Oil Storage Building (El. 242')	109
21. Emergency Service Water Electrical Equipment Room	104
22. Emergency Service Water Pump Room	122
23. 1A-SA & 1B-SB Exhaust Silencer Rooms (El. 292')	122
24. 1A-SA & 1B-SB H&V Equipment Rooms (El. 292')	122
25. 1A-SA & 1B-SB H&V Equipment Rooms (El. 280')	110
26. 1A-SA & 1B-SB Electrical Rooms (El. 261')	104
27. 1A-SA & 1B-SB Diesel Generator Rooms (El. 261')	122

3/4.7.~~X~~ 13 ESSENTIAL SERVICES CHILLED ~~SERVICE~~ WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~X~~ 13 At least two independent ~~service water~~ Essential Services Chilled Water System loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one ~~service water~~ Essential Services Chilled Water System loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

~~4.7.X 13 At least two service water Essential Services Chilled Water System loops shall be demonstrated OPERABLE:~~

~~a. At least once per 31 days by verifying that each manual valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.~~

~~b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation test signal.~~

4.7.13 No additional requirements other than those required by Specification 4.0.5

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Capitalize the Titles of the Following Systems and Component Names:

Diesel Generators and use for diesel or generator

Day Tanks and use for "Day and engine mounted tank"

Main Fuel Oil Storage Tank and use in place of "fuel storage tank"

Diesel Fuel Oil Transfer Pump and use in place of "fuel transfer pump"

Loss of Off-site Power

Safety Injection

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System, and

b. Two separate and independent diesel generators, each with:

1) ~~Separate~~ A day ~~and engine-mounted fuel~~ tanks containing a minimum volume of ~~2070~~ 2760 gallons of fuel WHICH IS EQUIVALENT TO 93% INDICATED LEVEL.

2) A separate MAIN Fuel OIL Storage ~~System~~ TANK containing a minimum volume of ~~75000~~ 83200 gallons of fuel WHICH IS EQUIVALENT TO __ % INDICATED LEVEL

3) A separate DIESEL fuel OIL transfer pump,

~~4) Lubricating oil storage containing a minimum total volume of 1100 gallons of lubricating oil, and~~

~~5) Capability to transfer lubricating oil from storage to the diesel generator unit~~ ~~VIA THE LUBE OIL KEEP WARM PUMP.~~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With ~~either an offsite~~ ONLY ONE PHYSICALLY INDEPENDENT OFFSITE circuit OPERABLE or EITHER OF THE REQUIRED Diesel Generators ~~of the above required A.C. electrical power sources~~ inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With ONLY one ~~offsite~~ PHYSICALLY INDEPENDENT OFFSITE circuit OPERABLE & and one Diesel Generator ~~of the above required A.C. electrical power sources~~ inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With ~~two of the above required offsite~~ ALL PHYSICALLY INDEPENDENT OFFSITE A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing the requirements of Specification 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing the requirements of Specification 4.8.1.1.1~~a~~ within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 ~~Each of the above required~~ TWO independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be~~:~~

~~a.~~ ~~D~~determined OPERABLE at least once per 7 days by verifying correct breaker alignments AND indicated power availability. ~~and~~

~~b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.~~

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1) Verifying the fuel level in the ~~d~~Day ~~and engine-mounted fuel~~ Tank,

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SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying the fuel level in the MAIN fuel OIL storage tank,

3) Verifying the DIESEL fuel OIL transfer pump starts and transfers fuel from the MAIN FUEL OIL storage ~~system~~ TANK to the Day ~~and engine mounted~~ tank,

~~4) Verifying the lubricating oil inventory in storage,~~

4 ~~5~~) Verifying the diesel starts from ambient condition and accelerates to at least ~~[900]~~ 450 rpm in less than or equal to ~~[10]~~ seconds.* The generator voltage and frequency shall be ~~[4160] ± [420]~~ 6900 ± 690 volts and [60] ± [1.2] Hz within [10] seconds* after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ~~ESF Actuation~~ SAFETY INJECTION test signal, or

d) An ~~ESF Actuation~~ SAFETY INJECTION test signal by itself.

5 ~~6~~) Verifying the generator ~~5754~~ 6500 is synchronized, loaded to greater than or equal to ~~[continuous rating]~~ kW in less than or equal to [60] seconds*, and operates with a load greater than or equal to ~~5754~~ 6500 ~~[continuous rating]~~ for at least 60 minutes, and

6 ~~7~~) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the Day ~~and engine-mounted fuel~~ tank~~s~~;

c. At least once per ~~92 days [or once per~~ 31 days ~~(if the groundwater table is equal to or higher than the bottom of the fuel oil storage tanks)]~~ by checking for and removing accumulated water from the fuel oil storage tanks;

d. By sampling new fuel oil in accordance with ASTM-D4057 prior to addition to storage tanks and:

1) By verifying in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks that the sample has:

---

*These diesel generator starts from ambient conditions shall be performed only once per 184 days in these surveillance tests and all other engine starts for the purpose of this surveillance testing shall be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

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SURVEILLANCE REQUIREMENTS (Continued)

a) An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes ~~(alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but not less than or equal to 40.1)~~, if gravity was not determined by comparison with the supplier's certification;

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

e. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;

f. At least once per 18 months, during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;

2) Verifying the generator capability to 1078 reject a load of greater than or equal to ~~[largest single emergency load]~~ kW while maintaining voltage at ~~[4160] ± [420]~~ volts and frequency at ~~[60] ± [1.2]~~ Hz; ~~[less than or equal to 75% of the difference between nominal speed and the Overspeed Trip Setpoint, or 15% above nominal whichever is less]~~; 6900 ± 690 60 ± 6.75 ~~LESS THAN 64.5~~

3) Verifying the generator capability to reject a load of ~~[continuous rating]~~ ~~5754~~ 6500 kW without tripping. The generator voltage shall not exceed ~~[4784]~~ ~~7590~~ 7935 volts during and following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected

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loads within [10] seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at ~~[4160] ± [420]~~ volts and [60] ± [1.2] Hz during this test. 6900±690

5) Verifying that on a ~~an ESF Actuation~~ SAFETY INJECTION test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 6900±690 ~~[4160] ± [420]~~ volts and [60] ± [1.2] Hz within [10] seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with a ~~an ESF Actuation~~ SAFETY INJECTION test signal, and:

a) Verifying deenergization of the emergency bus~~ses~~ (1A-SA OR 1B-SB) and REQUIRED load shedding from the emergency busses;

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within [10] seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at ~~[4160] ± [420]~~ 6900±690 volts and [60] ± [1.2] Hz during this test; and

c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, AND GENERATOR BUS FAULT are automatically bypassed upon loss of ~~voltage on the~~ OFFSITE POWER SIGNAL OR UPON ~~emergency bus concurrent with~~ a Safety Injection Actuation signal.

7) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to ~~[2-hour rating]~~ 7150 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to ~~[continuous rating]~~ 6500 kW. The generator voltage and frequency shall be ~~[4160] ± [420]~~ 6900±690 volts and [60] ± [1.2] Hz within [10] seconds after the start signal; the steady-state generator voltage and

frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2e.6)b);*

8) Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of ~~5754~~ 6500 kW;

9) Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal WITHOUT A LOSS OF OFFSITE POWER SIGNAL overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the SAFETY INJECTION ~~emergency~~ loads with offsite power;

~~11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day and engine-mounted tank of each diesel via the installed cross-connection lines;~~

11 ~~12~~) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval;

12 ~~13~~) Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

a) ~~[Turning gear engaged], or~~ OPERATIONAL AND MAINTENANCE SWITCH IN THE MAINTENANCE MODE

b) [Emergency stop].

~~13 14) Verifying that with all diesel generator air start receivers pressurized to less than or equal to ___ psig and the compressors secured, the diesel generator starts at least [5] times from ambient conditions and accelerates to at least [900] rpm in less than or equal to [10] seconds.~~

*If Specification 4.8.1.1.2e.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at ~~[continuous rating]~~ ~~5754~~ 6500 kW for 1 hour or until operating temperature has stabilized.

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SURVEILLANCE REQUIREMENTS (Continued)

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least ~~[900]~~ 450 rpm in less than or equal to [10] seconds; and

h. At least once per 10 years by:

1) Draining each MAIN Fuel Oil Storage Tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

2) Performing a pressure test of those portions of the Diesel GENERATOR Fuel Oil STORAGE AND TRANSFER System designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purpose of this schedule, only valid tests conducted after the completion of the preoperational test requirements of Regulatory Guide 1.108, Revision 1, August, 1977, shall be included in the computation of the "Last 100 Valid Tests."

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A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

1) Day ~~and engine mounted fuel~~ tank ~~s~~ containing a minimum volume of 2760 gallons of fuel WHICH IS EQUIVALENT TO 93% INDICATED LEVEL.

2) A MAIN fuel OIL storage ~~system~~ TANK containing a minimum volume of ~~75000~~ 83200 gallons of fuel WHICH IS EQUIVALENT TO % INDICATED LEVEL

3) A fuel OIL transfer pump,

~~4) Lubricating oil storage containing a minimum total volume of ____ gallons of lubricating oil, and~~

~~5) Capability to transfer lubricating oil from storage to the diesel generator unit.~~

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over ~~the~~ IRRADIATED ~~THE SPENT~~ fuel ~~storage pool~~, and within 8 hours, depressurize and vent the Reactor Coolant System through ~~a~~ greater than or equal to 2.54 square inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.6)), and 4.8.1.1.3.

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3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. ~~[250/~~125~~]~~-volt Battery Bank 1A-SA ~~No. 1~~, and EITHER ~~its associated~~ full capacity charger, ~~and~~ 1A-SA or 1B-SA, AND,

b. ~~[250/~~125]-volt Battery Bank 1B-SB ~~No. 2~~, and EITHER ~~its associated~~ full capacity charger, 1A-SB or 1B-SB.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

~~a.~~ With one of the required battery banks ~~and/or full capacity chargers~~ inoperable, restore the inoperable battery bank ~~and/or full capacity charger~~ to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~This specification is intended for use on plants with two divisions of D.C. power only. Modifications may be necessary, on a plant-unique basis, to accommodate different designs.~~

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each ~~[250/~~125~~]~~-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to ~~[250/~~129~~]~~ volts on float charge.

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b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below ~~[220/~~110~~]~~ volts, or battery overcharge with battery terminal voltage above ~~[300/~~150~~]~~ volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than ~~[~~150 x 10^{-6}~~]~~ ohm, and

3) The average electrolyte temperature of 10 ~~[a representative number]~~ of connected cells is above ~~[~~60~~]~~° F.

c. At least once per 18 months by verifying that:

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to ~~[~~150 x 10^{-6}~~]~~ ohm, and

4) The battery charger will supply at least 150 ~~[400]~~ amperes at GREATER THAN OR EQUAL TO ~~[~~125~~/250]~~ volts for at least ~~[8]~~ 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

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TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A[1]	CATEGORY B[2]	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE[3] VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[6]	> 2.07 volts
Specific Gravity[4]	≥ 1.200[5]	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195[5]

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than [2] amps when on charge.

(6) Corrected for average electrolyte temperature.

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ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

(EITHER 1A-SA OR 1B-SB)

3.8.2.2 As a minimum, one ~~[250/125]~~ volt battery bank and ~~the associated~~ AT LEAST ONE ASSOCIATED full-capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery bank and/or full-capacity charger inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required battery bank and full-capacity charger to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a 2.45 square inch vent.

b. With ~~the required~~ both full capacity chargers for the required safety related battery inoperable, demonstrate the OPERABILITY of its associated battery bank by performing the surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours there after. If any Category ~~B~~ A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required ~~[250/125]~~ volt SAFETY RELATED battery bank and full-capacity charger shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

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OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical bus~~s~~es shall be energized in the specified manner with tie breakers open ~~[both]~~ between redundant bus~~s~~es within the unit ~~[and between units at the same station]~~.

a. Division ~~#1~~ A ESF A.C. ~~Emergency~~ Bus~~s~~es consisting of:

1) ~~[4160]~~ 6900-Volt ~~Emergency~~ Bus # 1A-SA, and
2) [480]-Volt ~~Emergency~~ Bus # 1A2-SA.
3) 480 VOLT Bus # 1A3-SA

b. Division ~~#2~~ B ESF A.C. ~~Emergency~~ Bus~~s~~es consisting of:

1) ~~[4160]~~ 6900-Volt ~~Emergency~~ Bus # 1B-SB, and
2) [480]-Volt ~~Emergency~~ Bus # 1B2-SB.
3) 480 VOLT Bus # 1B3-SB

c. 118 ~~[120]~~-Volt A.C. Vital Bus # 1DP-1A-SI energized from its associated inverter connected to 125 VOLT D.C. Bus # __*__ DP-1A-SA

d. 118 ~~[120]~~-Volt A.C. Vital Bus # 1DP-1A-SII energized from its associated inverter connected to 125 VOLT D.C. Bus # __*__ DP-1A-SA

e. 118 ~~[120]~~-Volt A.C. Vital Bus # 1DP-1B-SIII energized from its associated inverter connected to 125 VOLT D.C. Bus # __*__ DP-1B-SB

f. 118 ~~[120]~~-Volt A.C. Vital Bus # 1DP-1B-SIV energized from its associated inverter connected to 125 VOLT D.C. Bus # __*__ DP-1B-SB

g. ~~[250/~~125]-Volt D.C. EMERGENCY Bus ~~#1~~ DP-1A-SA energized from EMERGENCY Battery ~~Bank #1~~ 1A-SA, and

h. ~~[250/~~125]-Volt D.C. EMERGENCY Bus ~~#2~~ DP-1B-SB energized from EMERGENCY Battery ~~Bank #2~~ 1B-SB.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required divisions of A.C. ~~emergency~~ ESF bus~~s~~es not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 118 VOLT A.C. vital bus either not energized from its associated inverter, or with the inverter 118 VOLT not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the 118 VOLT A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Two inverters may be disconnected from their D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their vital bus~~s~~es are energized, and (2) the vital bus~~s~~es associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With ~~one D.C. bus~~ EITHER 125V D.C. BUS 1A-SA OR 1B-SB not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical bus~~s~~es shall be energized in the specified manner:

Use Insertion following page.

a. One ~~division~~ ESF of A.C. ~~emergency~~ bus~~s~~es consisting of one ~~[4160]-volt~~ 6.9 KV and ~~one~~ [480]-volt A.C. ~~emergency~~ busES

b. 1B Two respective ~~[120]~~-volt A.C. uninterruptable ~~vital~~ ESF busses energized from their associated inverters connected to ~~their~~ respective D.C. bus~~s~~es, and

c. One ~~[250/125]~~ 125-volt D.C. bus (EITHER 1B-SB OR 1A-SA) energized from its associated battery bank.

APPLICABILITY MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible, and within 8 hours, depressurize and vent the RCS through at least a 2.45 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified bus~~s~~es shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the bus~~s~~es.

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SEE NEXT PAGE

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 AS A MINIMUM, ONE OF THE FOLLOWING DIVISIONS OF ELECTRICAL BUSES SHALL BE ENERGIZED IN THE SPECIFIED MANNER:

a. DIVISION A, CONSISTING OF:

1) 6.9 kV EMERGENCY BUS #1A-SA AND

2) 480 VOLT EMERGENCY BUSES #1A2-SA AND 1A3-SA, AND

3) 118-VOLT A.C. VITAL BUSES #1DP-1A-SI AND 1DP-1A-SIII ENERGIZED FROM THEIR ASSOCIATED INVERTER CONNECTED TO 125 V.D.C. BUS DP1A-SA, AND

4) 125 VOLT D.C. BUS #DP-1A-SA ENERGIZED FROM EMERGENCY BATTERY #1A-SA AND CHARGERS #1A-SA AND 1B-SA, OR

b. DIVISION B, CONSISTING OF:

1) 6.9 KV EMERGENCY BUS #1B-SB AND

2) 480 VOLT EMERGENCY BUSES #1B2-SB AND 1B3-SB, AND

3) 118 VOLT AC VITAL BUSES #1DP-1B-SII AND 1DP-1B-SIV ENERGIZED FROM THEIR ASSOCIATED INVERTER CONNECTED TO 125 VDC BUS-DP-1B-SB, AND

4) 125 VOLT D.C. BUS # DP-1B-SB ENERGIZED FROM EMERGENCY BATTERY #1B-SB AND CHARGERS #1B-SB, AND 1A-SB.

INSERT for page 8-16

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ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 At least the following A.C. circuits inside primary containment shall be deenergized:

a. Circuit numbers [___, ___, ___ and ___] in panel [ ].

b. Circuit numbers [___, ___, ___ and ___] in panel [ ].

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required A.C. circuits shall be determined to be deenergized at least once per 24 hours* by verifying that the associated circuit breakers are in the tripped condition.

*Except at least once per 31 days if locked, sealed, or otherwise secured in the tripped condition.

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CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.~~2~~ 1 All containment penetration conductor overcurrent protective devices given in ~~Table 3.8-1~~ FSAR TABLE 8.3.1-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) given in ~~Table 3.8-1~~ FSAR TABLE 8.3.1-10 inoperable:

a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.~~2~~ 1 All containment penetration conductor overcurrent protective devices given in ~~Table 3.8-1~~ FSAR TABLE 8.3.1-10 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1) By verifying that the ~~medium voltage [4-15 kV]~~ 6.9 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers ~~of each voltage level~~, and performing the following:

a) A CHANNEL CALIBRATION of the associated protective relays,

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed, and

SURVEILLANCE REQUIREMENTS (Continued)

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the pickup of the long-time delay trip element and 150% of the pickup of the short-time delay trip element, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ±20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested; and

3) By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a nondestructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	SYSTEM POWERED
1. 6900 VAC (Primary breaker) (Backup breaker)	Reactor Coolant pump 1 2 3 4
2. 480 VAC from MOAD Centers List all; primary breakers Backup breakers Backup breakers	
3. 480 VAC from MCC List all; primary breakers Backup breakers Backup breakers	
4. 125V DC Lighting List all; primary breakers Backup breakers Backup breakers	
5. 440 VAC CRDM Power Primary breakers Backup breakers Backup breakers	

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MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION ~~[Optional Bypassed]~~ AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

FSAR 8.3.1-11

3.8.4.~~3~~2 The thermal overload protection of each valve given in Table ~~3.8-2~~ shall be bypassed ~~[continuously] [or] [~~only under accident conditions~~] [, as applicable,]~~ by an OPERABLE bypass device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed under conditions for which it is designed to be bypassed, restore the inoperable device or provide a means to bypass the thermal overload within 8 hours, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.~~3~~2.1 The thermal overload protection for the above required valves shall be verified to be bypassed ~~[continuously] [or] [~~only under accident conditions~~] [, as applicable,]~~ by an OPERABLE integral bypass device ~~[verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing] [and] [or] [~~ by the performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions~~]~~:

a. At least once per ~~[~~18 months ~~for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing] [and] [or] [at least once per] [92 days~~ for those thermal overloads which are normally in force during plant operation and are bypassed only under accident conditions~~]~~; and

b. Following maintenance on the motor starter which has caused continuity in the bypassed circuitry to be interrupted (i.e. lifted leads, overload heater replacement, etc.)

~~[4.8.4.3.2 The thermal overload protection for the above required valves which are continuously bypassed and temporarily placed in force only when the valve motor is undergoing periodic or maintenance testing shall be verified to be bypassed following periodic or maintenance testing during which the thermal overload protection was temporarily placed in force.]~~

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION [Optional-Not Bypassed]

LIMITING CONDITION FOR OPERATION

3.8.4.3 The thermal overload protection of each valve given in Table 3.8-2 shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves inoperable, [continuously] bypass the inoperable thermal overload within 8 hours [; restore the inoperable thermal overload to OPERABLE status within 30 days] or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for the above required valves shall be demonstrated OPERABLE at least once per 18 months and following maintenance on the motor starter by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overloads for the above required valves.

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TABLE 3.8-2

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)(No)	SYSTEM(S) AFFECTED

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to [2000] ppm.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to [2000] ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 The Valves LISTED ON TABLE 4.9-1 ~~[Isolation of unborated water sources]~~ shall be verified ~~closed and~~ secured ~~in position by mechanical stops or by removal of air or electrical power at least once~~ in the positions required by Table 4.9-1 at least once per 31 days.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

TABLE 4.9-1

ADMINISTRATIVE CONTROLS
TO PREVENT DILUTION DURING REFUELING

Valve Location/ID	Valve Position During Refueling	Lock	Description
1CS-149	Closed	Yes	RMW to the CVCS makeup control system.
1CS-510	Closed	Yes	Boric Acid Batch Tank suction. Valve may be opened if the batching tank concentration is 2000 ppm boron, and valve 1-8302 (makeup water supply to batch tank) is closed.
1CS-503	Closed	Yes	Reactor Makeup Water to Batching Tank. Do not open unless suction valve 1-8308 is closed.
1CS-570	Closed	No	Place valve in "maintain close" at valve control switch and place BTRS master switch in "off." No lock required.
1CS-670	Closed	Yes	RMW to BTRS loop.
1CS-649	Closed	Yes	Resin sluice to BTRS demineralizers.
1CS-93	Closed	Yes	Resin sluice to CVCS deminerzlizers.
1CS-320	Closed	Yes	Recycle Evaporation Feed Pump to charging Pump Suction.
1CS-98	Open	No	BTRS isolation valve. Place valve control switch in "maintain open" position.

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3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ~~ANALOG~~ CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ~~ANALOG~~ CHANNEL OPERATIONAL TEST at least once per 7 days.

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3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least ~~100~~ 48 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than ~~100~~ 48 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 48 ~~100~~ hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

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3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

and Containment ~~&~~ Pre-Entry Purge

2) Be capable of being closed by an OPERABLE automatic Normal containment PURGE MAKEUP AND EXHAUST ~~VENTILATION SYSTEM~~ ~~purge and exhaust~~ isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

Purge and Containment ~~Pre-Entry~~ PURGE MAKEUP AND EXHAUST ~~VENTILATION SYSTEM~~

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic Normal containment ~~purge and exhaust~~ isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

Purge and Containment Pre-Entry PURGE MAKEUP AND EXHAUST ~~VENTILATION SYSTEM~~

b. Testing the Normal containment ~~purge and exhaust~~ isolation valves per the applicable portions of Specification 4.6.4.2.

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3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station IN THE CONTAINMENT BUILDING

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station IN THE CONTAINMENT BUILDING shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

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3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1) A minimum capacity of [2750] pounds, and

2) An overload cutoff limit less than or equal to [2700] pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

1) A minimum capacity of [610] pounds, and

2) A load indicator which shall be used to prevent lifting loads in excess of [600] pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least [2750] pounds and demonstrating an automatic load cutoff when the crane load exceeds [2700] pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least [610] pounds.

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3/4.9.6 REFUELING MACHINE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be operable with:

a. The refueling machine used for movement of fuel assemblies having:

1. A minimum capacity of 5000 pounds, and

2. Automatic overload cutoffs with the following setpoints:

   a. primary - 250 pounds above the indicated suspended weight for wet conditions and 100 pounds above the indicated suspended weight for dry conditions

   b) secondary - 150 pounds above the primary overload cutoff, and

3. An automatic load reduction trip with a setpoint of 250 pounds below the suspended weight for wet conditions and 350 pounds below the suspended weight for dry conditions.

b. The auxiliary hoist used for latching and unlatching drive rods and for thimble plug handling operations having:

1. A minimum capacity of 3000 pounds, and

2. A 1000 pound load indicator which shall be used to monitor lifting loads for these operations.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine and/or auxiliary hoist OPERABILITY not satisfied, suspend use of any inoperable refueling machine and/or auxiliary hoist from operations involving the movement of fuel asemblies and/or drive rods within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each refueling machine used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within

100 hours prior to start of such operations by performing a load test of at least 125% of the secondary automatic overload cutoff and by demonstrating an automatic load cutoff when the refueling machine load exceeds the setpoints of Specification 3.9.6.a.2.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 1250 pounds.

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3/4.9.7 CRANE TRAVEL - ~~SPENT FUEL STORAGE AREAS~~ FUEL HANDLING BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2300 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With IRRADIATED fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2300 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

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3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 WITH IRRADIATED FUEL IN THE VESSEL when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to ~~[2800]~~ 2500 gpm at least once per 12 hours.

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS AND CORE LOADING VERIFICATION in the vicinity of the reactor vessel hot legs.

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LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6, WITH IRRADIATED FUEL IN THE VESSEL when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours~~.~~

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to ~~[2800]~~ 2500 gpm at least once per 12 hours.

~~*Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.~~

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3/4.9.9 CONTAINMENT ~~PURGE AND EXHAUST~~ VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment ~~Purge and Exhaust~~ VENTILATION Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment ~~Purge and Exhaust~~ VENTILATION Isolation System inoperable, close each of the ~~purge~~ MAKEUP and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment ~~Purge and Exhaust~~ VENTILATION Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment ~~purge and exhaust~~ VENTILATION isolation occurs on ~~manual initiation and on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.~~ TWO-OUT-OF-FOUR HIGH RADIATION (REFER TO TABLE 3.3-6, ITEM 1.a) TEST SIGNAL FROM THE CONTAINMENT ATMOSHERE RADIATION MONITORS AND BY VERIFYING THAT EACH CONTAINMENT VENTILATION SYSTEM ISOLATION VALVE CAN BE CLOSED USING THE CONTROL SWITCH IN THE MAIN CONTROL ROOM

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3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

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3/4.9.11 WATER LEVEL - ~~STORAGE POOL~~ NEW AND SPENT FUEL POOLS.

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in ~~the storage~~ a pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the ~~fuel storage areas~~ AFFECTED POOL and restore the water level to within its limit within 4 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in ~~the storage~~ a pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the ~~fuel storage~~ a pool.

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3/4.9.12 FUEL ~~STORAGE POOL AIR CLEANUP SYSTEM~~ HANDLING BUILDING EMERGENCY EXHAUST

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel ~~Storage Pool Air Cleanup Systems~~ HANDLING BUILDING EMERGENCY EXHAUST SYSTEM TRAINS shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one Fuel ~~Storage Pool Air Cleanup System~~ HANDLING BUILDING EMERGENCY EXHAUST SYSTEM TRAIN inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel ~~Storage Pool Air Cleanup System~~ HANDLING BUILDING EMERGENCY EXHAUST SYSTEM TRAIN is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Fuel ~~Storage Pool Air Cleanup~~ HANDLING BUILDING EMERGENCY EXHAUST System TRAINS OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel ~~Storage Pool Air Cleanup~~ HANDLING BUILDING EMERGENCY EXHAUST System TRAIN is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel ~~Storage Pool Air Cleanup~~ HANDLING BUILDING EMERGENCY EXHAUST Systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

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SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[**]%~~ .05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the ~~system~~ unit flow rate is 6600 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~[**]%~~ 0.2%, and

3) Verifying a ~~system~~ unit flow rate of 6600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than ~~[**]%.~~ 0.2%

d. At least once per 18 months by:

1) Verifying that the TOTAL pressure drop across ~~the combined HEPA filters and charcoal adsorber banks is less than [6]~~ A FUEL HANDLING BUILDING EMERGENCY EXHAUST UNIT IS NOT GREATER THAN 9.27 inches Water Gauge while operating the ~~system~~ unit at a flow rate of 6600 acfm ± 10%,

2) Verifying that on a High Radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks,

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to ~~[1/4]~~ 1/8 inch Water Gauge relative to the outside atmosphere during system operation,

4) Verifying that the filter cooling bypass valve~~s~~ IS LOCKED ~~can be manually~~ opene~~d,~~ and

5) Verifying that the heaters dissipate 40 ± 4 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial UNIT replacement of a HEPA filter bank, by verifying that the ~~cleanup system~~ satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the ~~system~~ unit at a flow rate of 6600 cfm ± 10%.

f. After each complete or partial replacement UNIT of a charcoal adsorber bank, by verifying that the ~~cleanup system~~ satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[**]%~~ 0.5% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the ~~system~~ unit at a flow rate of 6600 cfm ± 10%.

~~*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% of less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).~~

~~**Value applicable will be determined by the following equation:~~

~~$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).~~

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated single control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any ~~full-length~~ control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all ~~full-length~~ control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each ~~full-length~~ control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each ~~full-length~~ control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

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3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to ~~[531]~~ 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than ~~[531]~~ 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ~~ANALOG~~ CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to ~~[531]~~ 541°F at least once per 30 minutes during PHYSICS TESTS.

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3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of STARTUP and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during STARTUP and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ~~ANALOG~~ CHANNEL OPERATIONAL TEST within 12 hours prior to initiating STARTUP and PHYSICS TESTS.

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SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual ~~full length~~ shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS ~~(see Figure 5.1-3)~~ 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 shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

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TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (µCi/ml)
1. Batch Waste Release Tanks(2)	P Each Batch	P Each Batch	Principal Gamma Emitters(3)	5×10^{-7}
a. WASTE MONITOR TANKS			I-131	1×10^{-6}
b. WASTE EVAPORATOR CONDENSATE TANKS	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite(4)	H-3	1×10^{-5}
c. SECONDARY WASTE SAMPLE TANKS			Gross Alpha	1×10^{-7}
d. TREATED LAUNDRY & HOT SHOWER STORAGE TANKS	P Each Batch	Q Composite(4)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Continuous Releases(5)(7)	~~Continuous(6)~~	~~W Composite(6)~~	~~Principal Gamma Emitters(3)~~	~~5×10^{-7}~~
			~~I-131~~	~~1×10^{-6}~~
a. NORMAL SERVICE WATER RETURN TO COOLING TOWER (POTENTIAL CONTINUOUS RELEASE)	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
~~b.~~	Continuous(6)	M Composite(6)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
~~c.~~	~~Continuous(6)~~	~~Q Composite(6)~~	~~Sr-89, Sr-90~~	~~5×10^{-8}~~
			~~Fe-55~~	~~1×10^{-6}~~

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TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^{-6} = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

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.

(2) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

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TABLE 4.11-1 (Continued)

TABLE NOTATIONS (Continued)

(3) The principal gamma emmiters for which the LLD specification applies include 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 also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.47 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(4) 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 that is representative of the liquids released.

(5) A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

(6) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

(7) THE SERVICE WATER LIQUID RELEASE REPRESENTS A POTENTIAL RELEASE PATHWAY AND NOT AN ACTUAL RELEASE PATHWAY. SURVEILLANCE OF THIS PATHWAY IS INTENDED TO ALERT THE PLANT TO A POTENTIAL PROBLEM; ANALYSIS FOR PRINCIPAL GAMMA EMITTERS SHOULD BE SUFFICIENT TO MEET THIS INTENT. IF ANALYSIS FOR PRINCIPAL GAMMA EMITTERS INDICATES A PROBLEM (i.e. EXCEEDS THE TRIGGER LEVEL OF (LATER) μCi/ml), THEN ANALYSES WILL BE PERFORMED ON A GRAB SAMPLE FOR I-131, PRINCIPAL GAMMA EMITTERS, H-3, GROSS ALPHA, Sr-89, Sr-90, AND Fe-55 USING THE LLD AS SPECIFIED IN TABLE 4.11-1 FOR BATCH RELEASES.

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RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. 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, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. ~~This Special Report shall also include: (1) the results of radiological analyses of the drinking water source, and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141, Safe Drinking Water Act.*~~

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

~~*The requirements of ACTION a.(1) and (2) are applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge. In the case of river-sited plants this is 3 miles downstream only.~~

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LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, ~~from each unit~~, to UNRESTRICTED AREAS (see Figure 5.1-~~4~~ 3) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that 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, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

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RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases:

~~a.~~

~~b.~~

~~c.~~

a. ~~d.~~ Outside temporary tank, EXCLUDING DEMINERALIZER VESSELS AND LINER USED TO SOLIDIFY OR DEWATER RADIOACTIVE WASTES.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.~~X~~7.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

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RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-~~3~~ 1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The 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 above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

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TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (μCi/ml)
1. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	1×10^{-4}
2. Containment Purge ~~or Vent~~	P Each PURGE(3) Grab Sample	P Each PURGE(3)	Principal Gamma Emitters(2)	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
3. VENTS a. Plant Vent STACK b. TURBINE BUILDING VENT STACK c. WASTE PROCESSING BUILDING VENT STACK 5	M(3),(4),(5) Grab Sample		Principal Gamma Emitters(2)	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
~~b. Fuel Storage Area Ventilation~~	~~M(5)~~ ~~Grab Sample~~		~~Principal Gamma Emitters(2)~~	~~1x10-4~~
			~~H-3 (oxide)~~	~~1x10-6~~
d. WASTE PROCESSING BUILDING VENT STACK 5A ~~e. Auxiliary Bldg, Radwaste Area, SGB Vent, Others~~	M Grab Sample	M	Principal Gamma Emitters(2)	1×10^{-4}
4. All Release Types as listed in 1., 2., and 3. above	Continuous(6)	W(7) Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous(6)	W(7) Particulate Sample	Principal Gamma Emitters(2)	1×10^{-11}
	Continuous(6)	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(6)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS

(1)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\, s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate. (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^{-6} = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

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.

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

(2) The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, CS-137, Ce-141 and Ce-144 in Iodine and particulate releases. 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 also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 in the format outlined in Regulatory Guide 1.2, Appendix B, Revision 1, June 1974.

(3) Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period.

(4) Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

(5) Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

(6) 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 Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(7) Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

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RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. 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, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit the the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the ~~WASTE GAS HOLDUP~~ GASEOUS RADWASTE TREATMENT SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that 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, and

3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

4.11.2.4.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM and ~~WASTE GAS HOLDUP~~ GASEOUS RADWASTE TREATMENT SYSTEM shall be considered OPERABLE by meeting Specifications 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

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EXPLOSIVE GAS MIXTURE ~~[Systems not designed to withstand a hydrogen explosion]~~

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the ~~WASTE GAS HOLDUP~~ GASEOUS RADWASTE TREATMENT SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the ~~WASTE GAS HOLDUP~~ GASEOUS RADWASTE TREATMENT SYSTEM greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the ~~WASTE GAS HOLDUP~~ GASEOUS RADWASTE TREATMENT SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the ~~WASTE GAS HOLDUP~~ GASEOUS RADWASTE TREATMENT SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the ~~WASTE GAS HOLDUP~~ GASEOUS RADWASTE TREATMENT SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

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RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE [Systems designed to withstand a hydrogen explosion]

LIMITNG CONDITION FOR OPERATION

3.11.2.5 The concentration of hydrogen or oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of hydrogen or oxygen in the WASTE GAS HOLDUP SYSTEM exceeding the limit, restore the concentration to within the limit within 48 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen or oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen or oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

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RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 1.05x10^5 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

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RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be solidified or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures, and/or the Solid Waste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM;

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste; and

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

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RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations shall be made including direct radiation contributions from the units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405(c), shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.~~3~~6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. 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 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a 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 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 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 from all radionuclides is equal to or greater than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, or 3.11.2.3. 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 Operating Report required by Specification 6.9.1.~~3~~6.

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

#STS
Shearon Harris Unit 1

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM*

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation(2)	Forty routine monitoring stations ~~(DR1-DR40)~~ either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY ~~(DR1-DR16)~~; An outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site ~~(DR17-DR32)~~; and The balance of the stations ~~(DR33-DR40)~~ to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly.	Gamma dose quarterly.

*The number, media, frequency, and location of samples may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an individual's dose and should be included in the sample program. The code letters in parentheses, e.g., DR1, A1, provide one way of defining sample locations in this specification that can be used to identify the specific locations in the map(s) and table in the ODCM.

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~~#STS~~ SHEARON HARRIS UNIT 1

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from five locations ~~(A1-A5)~~. Three samples ~~(A1-A3)~~ from close to the three SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground-level D/Q; One sample ~~(A4)~~ from the vicinity of a community having the highest calculated annual average ground-level D/Q; and One sample ~~(A5)~~ from a control location, as for example 15 to 30 km distant and in the least prevalent wind direction.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[3] and gamma isotopic analysis[4] of composite (by location) quarterly.
3. Waterborne			
a. Surface[5]	One sample upstream ~~(Wa1)~~. One sample downstream ~~(Wa2)~~.	Composite sample over 1-month period.[6]	Gamma isotopic analysis[4] monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from one or two sources ~~(Wb1, Wb2)~~, only if likely to be affected[7].	Quarterly.	Gamma isotopic[4] and tritium analysis quarterly.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne (Continued)	ONE SAMPLE IN THE VICINITY OF THE NEAREST DOWNSTREAM MUNICIPAL WATER SUPPLY INTAKE FROM THE CAPE FEAR RIVER		
c. Drinking	~~One sample of each of one to three (Wc1 - Wc3) of the nearest water supplies that could be affected by its discharge.~~ One sample from a control location (Wc4).	Composite sample over 2-week period(6) when I-131 analysis is performed; monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year(8). Composite for gross beta and gamma isotopic analyses(4) monthly. Composite for tritium analysis quarterly.
	ONE SAMPLE IN THE VICINITY OF THE COOLING TOWER BLOWDOWN DISCHARGE IN AN AREA WITH EXISTING OR POTENTIAL RECREATIONAL VALUE.		
d. Sediment from Shoreline	~~One sample from downstream area with existing or potential recreational value (Wd1).~~	Semiannually.	Gamma isotopic analysis(4) semiannually.
4. Ingestion			
a. Milk	Samples from milking animals in three locations ~~(Ia1 - Ia3)~~ within 5 km distance having the highest dose potential. If there are none, then one sample from milking animals in each of three areas ~~(Ia1 - Ia3)~~ between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr.(8) One sample from milking animals at a control location (Ia4), 15 to 30 km distant and in the least prevalent wind direction.	Semimonthly when animals are on pasture; monthly at other times.	Gamma isotopic(4) and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

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SHEARON HARRIS UNIT 1

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Continued)			
b. Fish and Invertebrates	One sample of ~~each commercially and recreationally important~~ BLUEGILLS, LARGE-MOUTH BASS AND CATFISH species in the vicinity of plant discharge area. ~~(Ib1 - Ib_)~~. One sample of same species in areas not influenced by plant discharge ~~(Ib10 - Ib_)~~.	Sample in season, or semiannually if they are not seasonal.	Gamma isotopic analysis(4) on edible portions.
c. Food Products	~~One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged (Ic1 - Ic_).~~	~~At time of harvest(9)~~.	~~Gamma isotopic analyses(4) on edible portion.~~
	Samples of three different kinds of broad leaf FOOD CROP vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q if milk sampling is not performed ~~(Ic10 - Ic13)~~.	Monthly during HARVEST ~~growing~~ season~~s~~ AS AVAILABLE	Gamma isotopic(4) and I-131 analysis.
	One sample of each of the similar broad leaf FOOD CROP vegetation grown 15 to 30 km distant in the least prevalent wind direction if milk sampling is not performed ~~(Ic20 - Ic23)~~.	Monthly during HARVEST ~~growing~~ season~~s~~ AS AVAILABLE	Gamma isotopic(4) and I-131 analysis.

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TABLE 3.12-1 (Continued)

TABLE NOTATIONS

(1) Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable specific alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program given in the ODCM. Pursuant to Specification 6.14 submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the selection of the new location(s) for obtaining samples.

(2) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. (The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.)

(3) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

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TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

(4) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(5) The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.

(6) A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

(7) Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

(8) The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

(9) If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

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TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

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TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[1] [2]

LOWER LIMIT OF DETECTION (LLD)[3]

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[4]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

*If no drinking water pathway exists, a value of 3000 pCi/l may be used.

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TABLE 4.12-1 (Continued)

TABLE NOTATIONS

(1) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3. 6

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (picoCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between environmental collection, or end of the sample collection period, and time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

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TABLE 4.12-1 (Continued)

TABLE NOTATIONS (Continued)

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. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

(4) LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

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RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 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.]

APPLICABILITY: At all times.

ACTION:

a. 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 4.11.2.3, pursuant to Specification 6.9.1.4? identify the new location(s) in the next Semiannual Radioactive Effluent Release Report.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. 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. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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*Broad leaf FOOD CROP vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1, Part 4.c., shall be followed, including analysis of control samples.

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SURVEILLANCE REQUIREMENTS

4.12.2 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 Operating Report pursuant to Specification 6.9.1.~~3~~6.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.~~3~~6.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.~~3~~6.

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3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. In the event of a disagreement between the requirements stated in these Technical Specifications and those stated in an applicable Federal Regulation or Act, the requirements stated in the applicable Federal Regulation or Act shall take precedence and shall be met.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 The specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.5.2 requires two independent ECCS subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS subsystem is inoperable. Under the requirements of Specification 3.0.3, if both the required ECCS subsystems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one Spray System is inoperable. Under the requirements of Specification 3.0.3, if both the required Containment Spray Systems are inoperable, within 1 ho measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours. It is acceptable to initiate and complete a reduction in OPERATIONAL MODES in a shorter time interval than required in the ACTION statement and to add the unused portion of this allowable out-of-service time to that provided for operation in subsequent lower OPERATION MODE(S). Stated allowable out-of-service times are applicable regardless of the OPERATIONAL MODE(S) in which the inoperability is discovered but the times provided for achieving a mode reduction are not applicable if the inoperability is discovered in a mode lower than the applicable mode. For example if the Containment Spray System was discovered to be inoperable while in STARTUP, the ACTION Statement would allow up to 156 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 78 hours, 140 hours would still be available before the plant would be required to be in COLD SHUTDOWN However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provided to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 156 hours to 150 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with: (1) the full complement of required systems, equipment, or components OPERABLE and (2) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

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APPLICABILITY

BASES

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under these criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE when such items are found or known to be inoperable although still meeting the Surveillance Requirements. Items may be determined inoperable during use, during surveillance tests, or in accordance with this specification. Therefore, ACTION statements are entered when the Surveillance Requirements should have been performed rather than at the time it is discovered that the tests were not performed.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant STARTUP or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

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BASES

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and 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 intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and 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, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

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3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of ~~[1.6%] Δk/k~~ 1770 pcm is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal, ~~and~~ but a ~~1% Δk/k~~ 2000pcm SHUTDOWN MARGIN is required to provide~~s~~ adequate protection for postulated inadvertant dilution events.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions~~;~~. Accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

THE POSITIVE LIMIT IS BASED ON CORE CONDITIONS FOR ALL RODS WITHDRAWN BEGINNING OF CYCLE, 0% THERMAL POWER; THE NEGATIVE LIMIT IS BASED ON CORE CONDITIONS FOR ALL RODS WITHDRAWN, END OF CYCLE, RATED THERMAL POWER.

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MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value ~~[-3.9] x 10-4 Δk/k/°F.~~ -42 pcm/°F The MTC value of ~~[-3.0] x 10-4 Δk/k/°F~~ -33 pcm/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of ~~[-3.9] x 10-4 Δk/k/°F.~~ -42 pcm/°F

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than ~~[541]~~ 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, ~~(3) the P-12 interlock is above its setpoint,~~ ~~(4)~~ 3 the pressurizer is capable of being in an OPERABLE status with a steam bubble, and ~~(5)~~ 4 the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated Heat Tracing Systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN

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BASES

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of ~~1.6% Δk/k~~ 1770 pcm after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 16300 ~~[5106]~~ gallons of [7000] ppm borated water be maintained in ~~from~~ the boric acid storage tanks or [~~52,622~~] gallons of 2000 ppm borated water be maintained in ~~from~~ the refueling water storage tank (RWST).

LATER

With the RCS temperature below 200°F, one boron injection ~~System~~ flow path is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection ~~System~~ flow path becomes inoperable.

The limitation for a maximum of one ~~centrifugal~~ charging/safety injection (CSIP) pump to be OPERABLE and the Surveillance Requirement to verify all ~~charging pumps~~ CSIPs except the required OPERABLE pump to be inoperable below ~~[275]~~ 250°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of ~~1% Δk/k~~ 1000 pcm after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5400 gallons of ~~[7000]~~ ppm borated water be maintained ~~from~~ in the boric acid storage tanks or (LATER) gallons of 2000 ppm borated water be maintained ~~from~~ in the RWST.

~~The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.~~

INSERT next page

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between ~~[8.5]~~ and ~~[11.0]~~ LATER for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. ~~Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120,~~

INSERT FOR PAGE B 3/4 1-3

B 3/4.1.2

The gallons given above are the amounts that need to be ~~taken~~ maintained in the ~~from the~~ tank in the various circumstances. To get the specified value each value had added to it (1) the unusable volume of water in the tank (2200 gal. for the BAT, 35460 gallons for RWST) and a 3% allowance for possible instrument error (1018 gallons for the BAT and 13900 gallons for the RWST). In addition, for human factors purposes the percent indicated levels were then raised to the next whole percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values and the specified % level and gallons differ by less than .1%.

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BASES

MOVABLE CONTROL ASSEMBLIES (Continued)

~~and 228 steps withdrawn for the Control Banks and 18, 210, and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.~~

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551 ~~[541]~~°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of ~~2.32~~ 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference (TARGET AFD) is determined at equilibrium xenon conditions. The ~~full-length~~ rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

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AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

90% Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than ~~90%~~ of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and ~~90%~~ and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

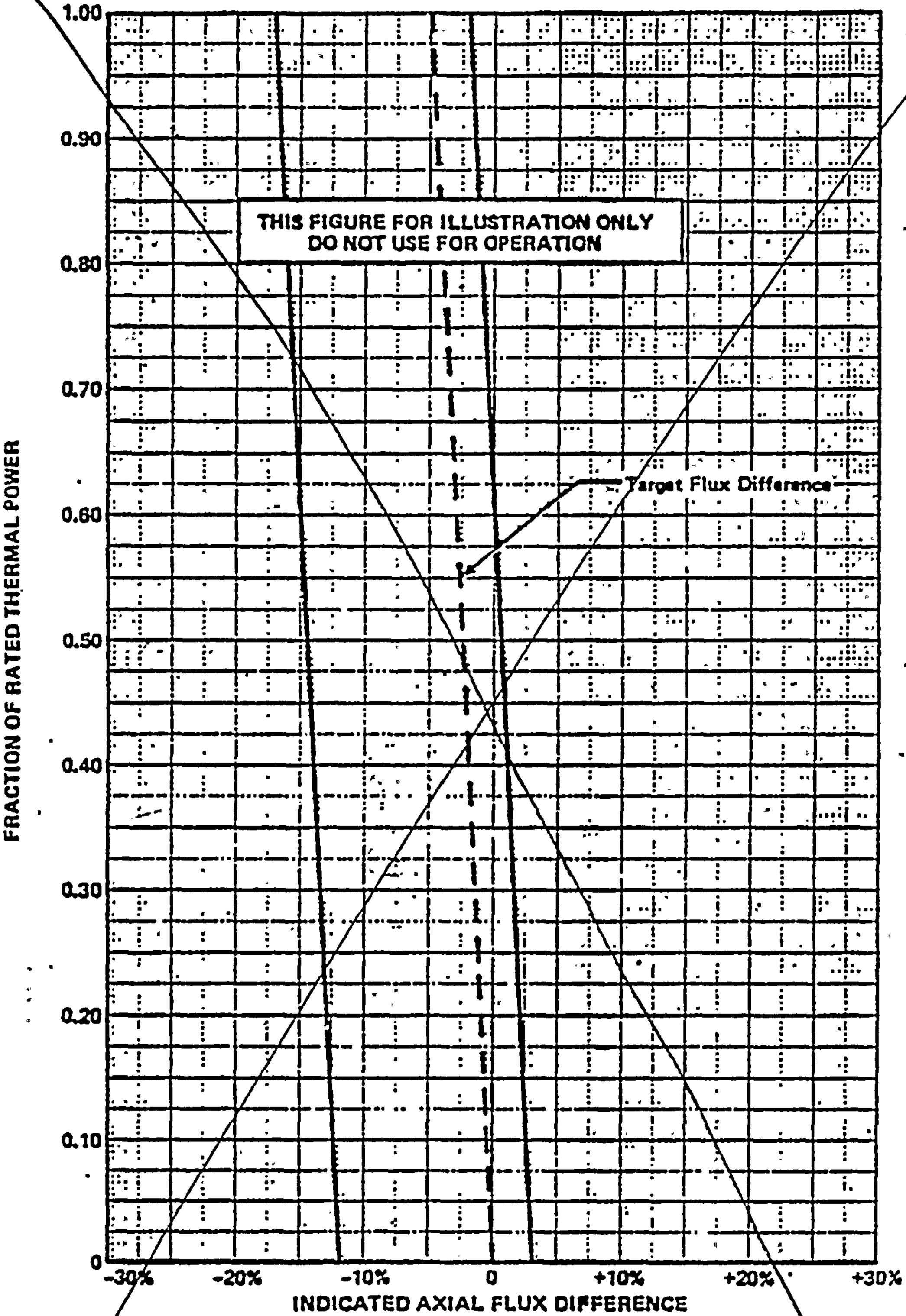

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

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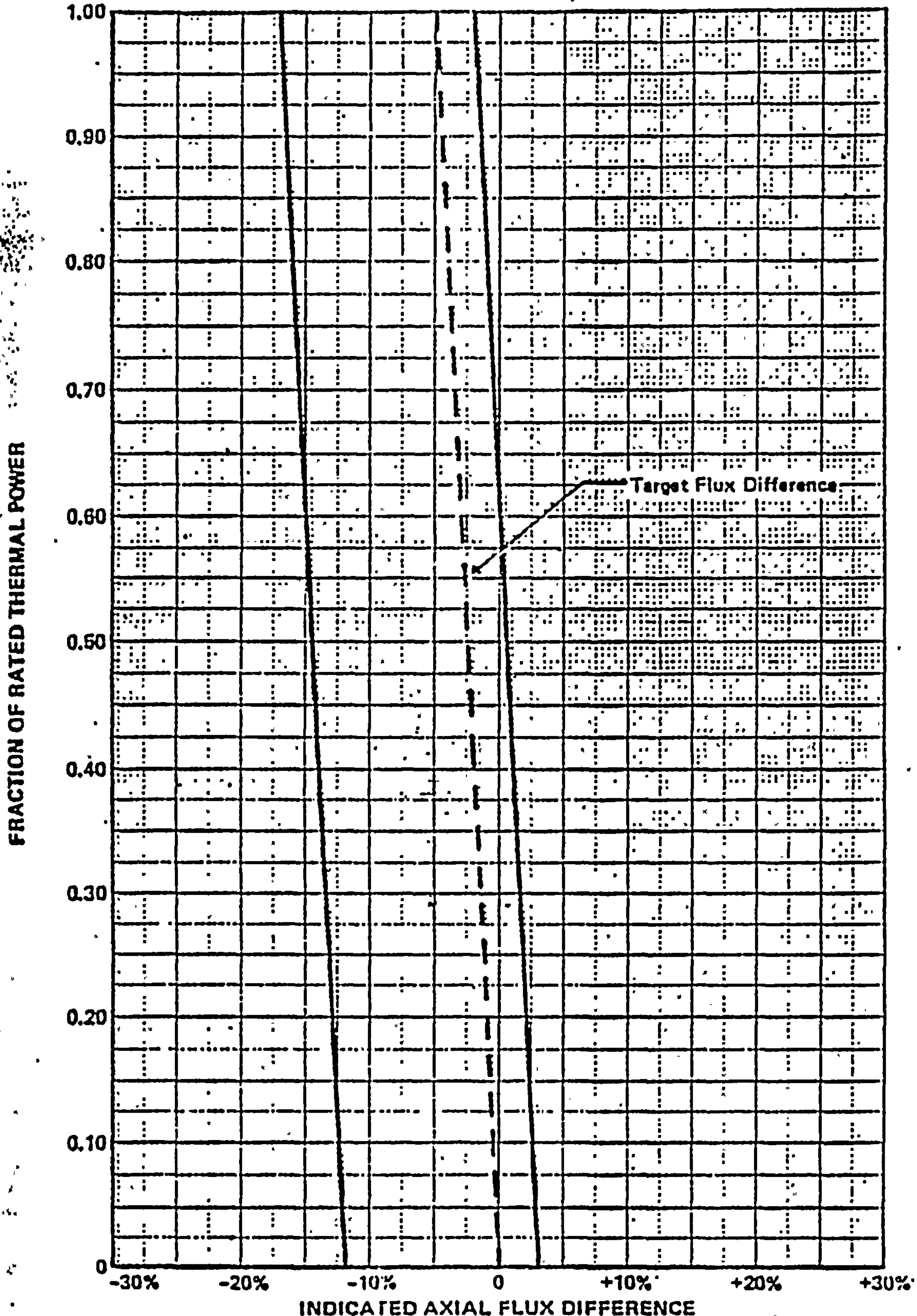

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FIGURE B 3/4.2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER FOR BURNUP GREATER THAN 3000 MWD/MTU

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. This margin includes the following:

a. Design limit DNBR of [1.30 vs 1.28],

b. Grid Spacing (K_s) of [0.046 vs 0.059],

c. Thermal Diffusion Coefficient of [0.038 vs 0.059],

d. DNBR Multiplier of [0.86 vs 0.88], and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

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HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

REFER TO NEXT PAGE FOR INSERT

~~When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and~~ a ~~3% allowance is appropriate for manufacturing tolerance.~~

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^{N}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 ~~and 3.2-4~~. Measurement errors of ~~[2.1]~~ 2.4% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. ~~Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of [0.1]% for undetected fouling of the feedwater venturi is included in Figure 3.2-3. Any fouling which might bias the RCS flow rate measurement greater than [0.1]% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall~~ be ~~cleaned to eliminate the fouling.~~

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation. ~~shown on Figure 3.2-3~~.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt 1.025 can be tolerated before the margin for uncertainty in FQ is depleted. A limit of 1.02 ~~limit of 1.02~~ was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

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When an FQ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate error allowance for a full core map taken with the incore detector flux mapping system and 3 percent is the appropriate allowance for manufacturing tolerance. This error and tolerance have been included in the uncertainty analysis for determination of the limiting value of FQ and need not be included in the measured value of FQ.

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BASES

QUADRANT POWER TILT RATIO (Continued)

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. ~~The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.~~

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of ~~[581]~~ LATER °F and the indicated pressurizer pressure value of ~~[2220]~~ LATER psig correspond to analytical limits of ~~595°F and 2205~~ 592.6°F 2205 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

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3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy. INSERT next page

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of

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noft For example, if a bistable has a trip setpoint of ≤ 100%, has a span of 125%, and has a calibration accuracy of ± 0.50%, then the bistable is considered to be adjusted to the trip setpoint as long as the "as measured" value for the bistable is ≤ 100.62%.

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REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for ~~REPORTABLE EVENTS~~, determination of OPERABILITY.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time.

The Engineered Safety Features Acutation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) CHARGING/Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment ~~cooling~~ fans COOLERS start and automatic valves position, (11) ~~essential~~ service water pumps start and automatic valves position, and (12) Control EMERGENCY Room Isolation and ~~Ventilation Systems~~ EMERGENCY FILTRATION start.

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REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure. ~~On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure.~~ INSERT following page.

~~P-12 On increasing reactor coolant loop temperature, P-12 automatically reinstates Safety Injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure, and provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 allows the manual block of Safety Injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure and automatically removes the arming signal from the Steam Dump System.~~

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~~P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves and inhibits feedwater control valve modulation.~~

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency ~~Exhaust or Ventilation~~ Systems.

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~~INSERT U~~ - INSTRUMENTATION

and Steam Line Pressure - Low and automatically blocks Steam Line Isolation on Negative Steam Line Pressure Rate-High. On decreasing pressure, P-11 allows the manual block of Safety Injection on Pressurizer Pressure-Low and Steam Line Pressure-Low and allows Steam Line Isolation on Negative Steam Line Pressure Rate-High to become active upon manual block of Steam Line Pressure-Low Safety Injection.

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3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 ~~ALTERNATE AND~~ REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, REMOTE SHUTDOWN SYSTEM (Continued)

control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection Systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, Revision 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

The OPERABILITY of the fire detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility Fire Protection Program.

~~Fire detectors that are used to actuate Fire Suppression Systems represent a more critically important component of a plant's Fire Protection Program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.~~

~~The loss of detection capability for Fire Suppression Systems, actuated by fire detectors, represents a significant degradation of fire protection for~~

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In the event that a portion of the fire detection instrumentation

FIRE DETECTION INSTRUMENTATION (Continued) is inoperable,

~~any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning.~~ the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the Loose-Part Detection System ensures that sufficient capability is available to detect loose metallic parts in the Reactor System and avoid or mitigate damage to Reactor System components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING 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 and adjusted in accordance with the methodology and parameters in the 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. ~~The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.~~

3/4.3.3.11 RADIOACTIVE GASEOUS 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 and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the ~~WASTE GAS HOLDUP 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. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1×10^{-6} µCi/cc are measurable.

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3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment, or structures.

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3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 250 ~~[275]~~°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G ~~by either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2)~~ by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50 °F above each of the RCS cold leg temperatures.

[OPTIONAL]

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its stop valves ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

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REACTOR COOLANT LOOPS AND COOLANT CIRCULATION (Continued)

[OPTIONAL]

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water-induced reactivity transient.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the ~~first~~ SECOND Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all other tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

~~PLUGGING WILL BE REQUIRED FOR TUBES IN THE PREHEATER SECTION WITH IMPERFECTIONS EXCEEDING THE PLUGGING LIMIT OF 40 % OF THE TUBE NOMINAL WALL THICKNESS~~

STEAM GENERATORS (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be ~~promptly~~ reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators ~~not isolated from the RCS~~ ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds ~~40~~ 31 gpm with the modulating

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OPERATIONAL LEAKAGE (Continued)

valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an

SPECIFIC ACTIVITY (Continued)

appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the SHEARON HARRIS site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1 microCurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube rupture.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radio-iodine contribution would probably be about 20%. The exclusion of radio-nuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

SPECIFIC ACTIVITY (Continued)

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

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PRESSURE/TEMPERATURE LIMITS (Continued)

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than [625]°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

WERE PERFORMED The fracture toughness TESTING ~~properties~~ of the ferritic materials in the reactor vessel ~~are determined~~ in accordance with the ~~NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements.~~ These properties are then evaluated in accordance with ~~Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.~~ THE NRC STANDARD REVIEW PLAN

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of ~~[12]~~ 5 effective full power years (EFPY) of service life. The ~~[12]~~ 5 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. LATER The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of ~~[12]~~ 5 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

1971 WINDER ADDENDA TO SECTION III OF THE ASME BOILER AND PRESSURE VESSEL CODE.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	ASME MATERIAL TYPE	CU %	P %	NDTT °F	50 FT-LB/35 MIL TEMP °F		RT_{NDT} °F	MIN. UPPER SHELF FT-LB	
						LONG	TRANS		LONG	TRANS

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REACTOR VESSEL TOUGHNESS

COMPONENT	GRADE	HEAT NO	CU %	P (%)	T_{NDT} (^{o}F)	RT_{NDT} (^{o}F)	AVG. SHELF ENERGY MWD FT-LB	AVG. SHELF ENERGY NMWD FT-LB
Closure Hd. Dome	A533,B,CL1	A9213-1	-	-	-10	8	-	114
Head Flange	A508,CL2	5302-V2	-	-	0	0	-	135
Vessel Flange	"	5302-V1	-	-	-10	-8	-	110
Inlet Nozzle	"	4388-4	-	-	-20	-20	-	169
" "	"	4388-5	-	-	0	0	-	128
" "	"	4388-6	-	-	-20	-20	-	149
Outlet Nozzle	"	4398-4	-	-	-10	-10	-	151
" "	"	4398-5	-	-	-10	-10	-	152
" "	"	4398-6	-	-	-10	-10	-	150
Nozzle Shell	A533B,CL1	C0224-1	.12	.008	-20	-1	-	90
" "	"	C0123-1	.12	.006	0	42	-	84
Inter. Shell	"	A9153-1	.09	.007	-10	60	106	83
" "	"	B4197-2	.10	.006	-10	90	112	74
Lower Shell	"	C9924-1	.08	.005	-10	54	147	98
" "	"	C9924-2	.08	.005	-20	57	148	88
Bottom Hd. Torus	"	A9249-2	-	-	-40	14	-	94
" " Dome		A9213-2	-	-	-40	-8	-	125
Weld (Inter & Lower Shell Vertical Weld Seams)			.06	.013	-20	-20	-	>94
Weld (Inter. to Lower Shell Girth Seam)			.04	.013	-20	-20	-	88

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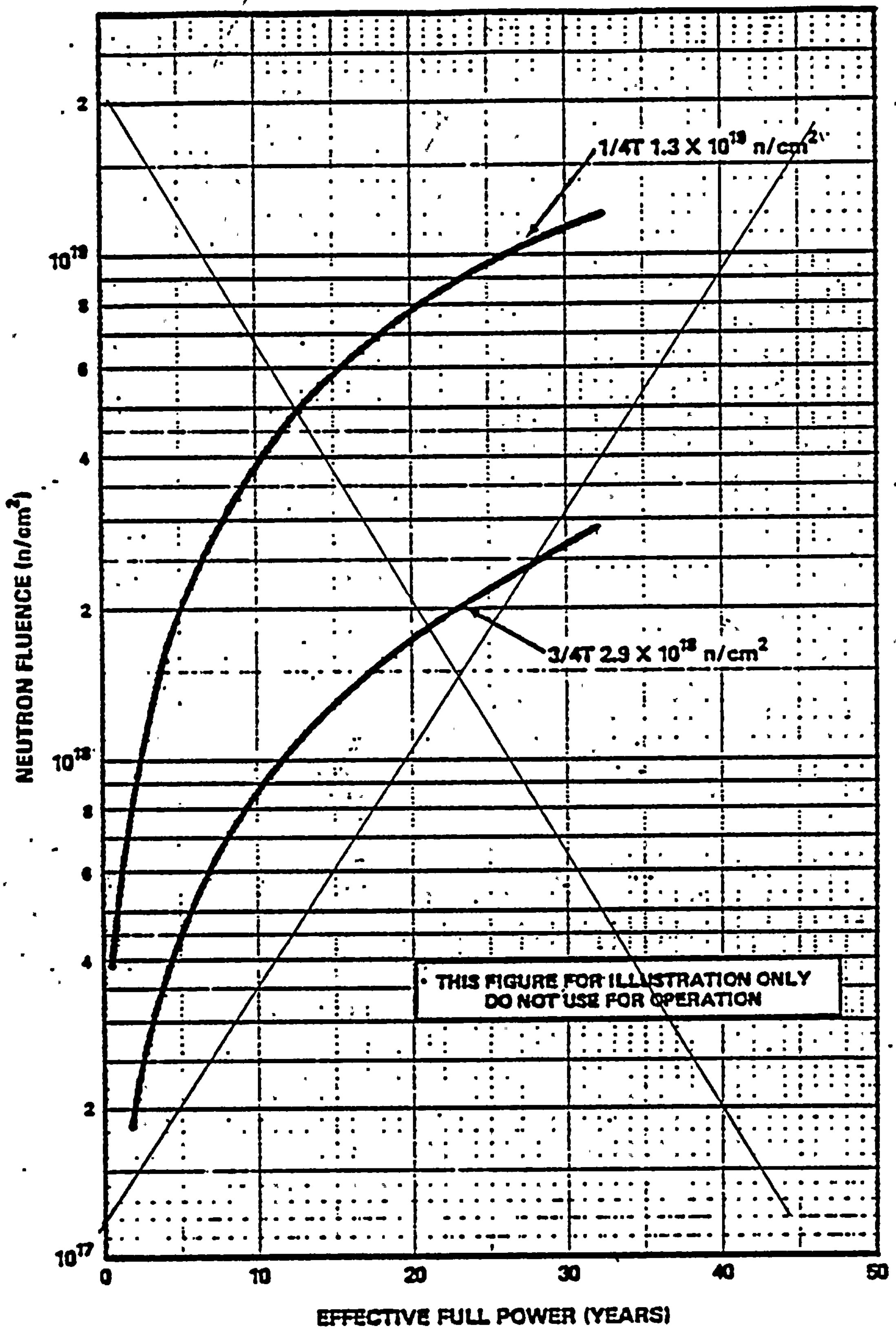

FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

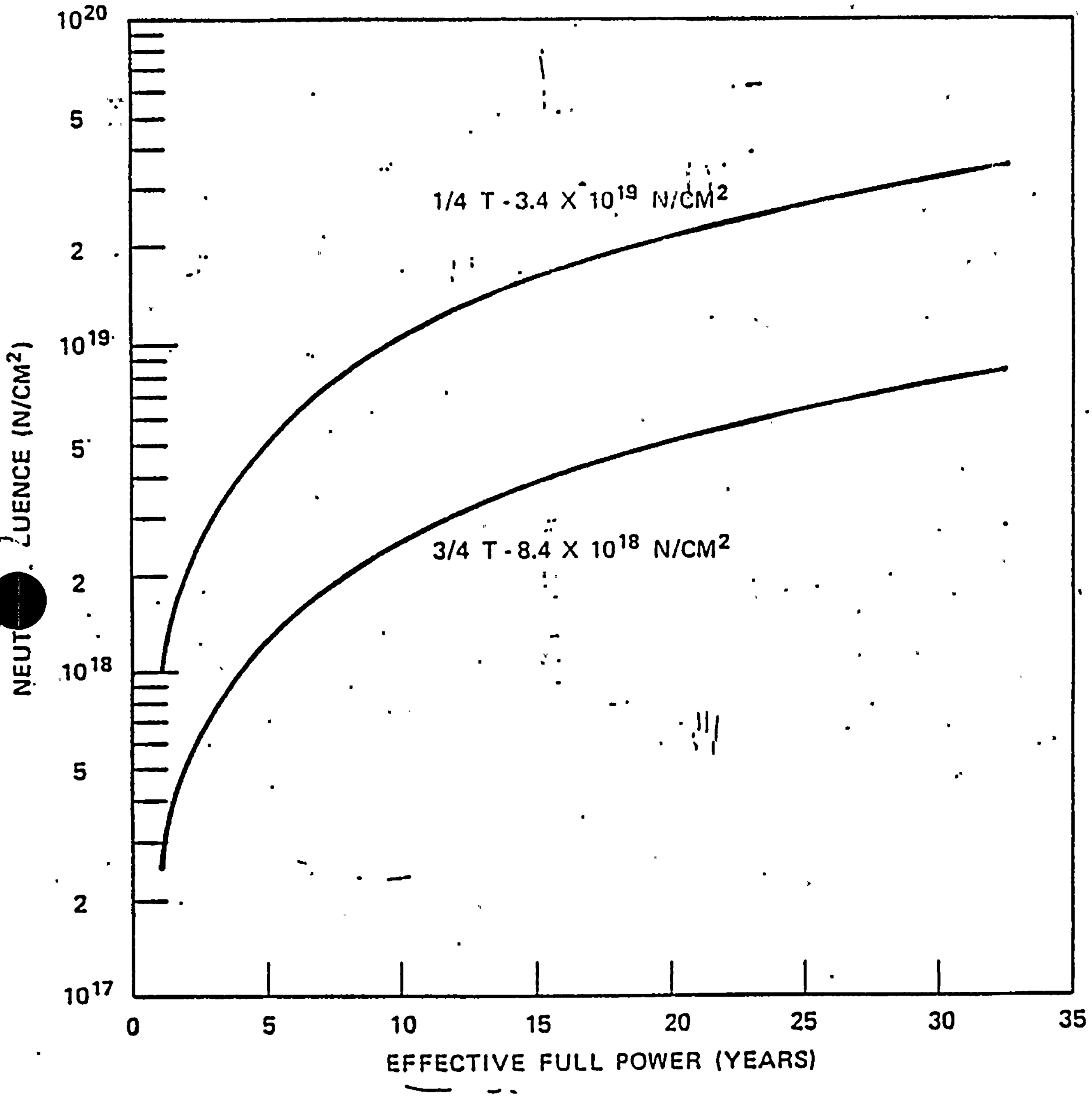

Figure B 3/4 4.1. Fast Neutron Fluence (E > 1 MeV) as a Function of Full Power Service Life

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SHEARON HARRIS-UNIT 1
~~CALLAWAY - UNIT 1~~

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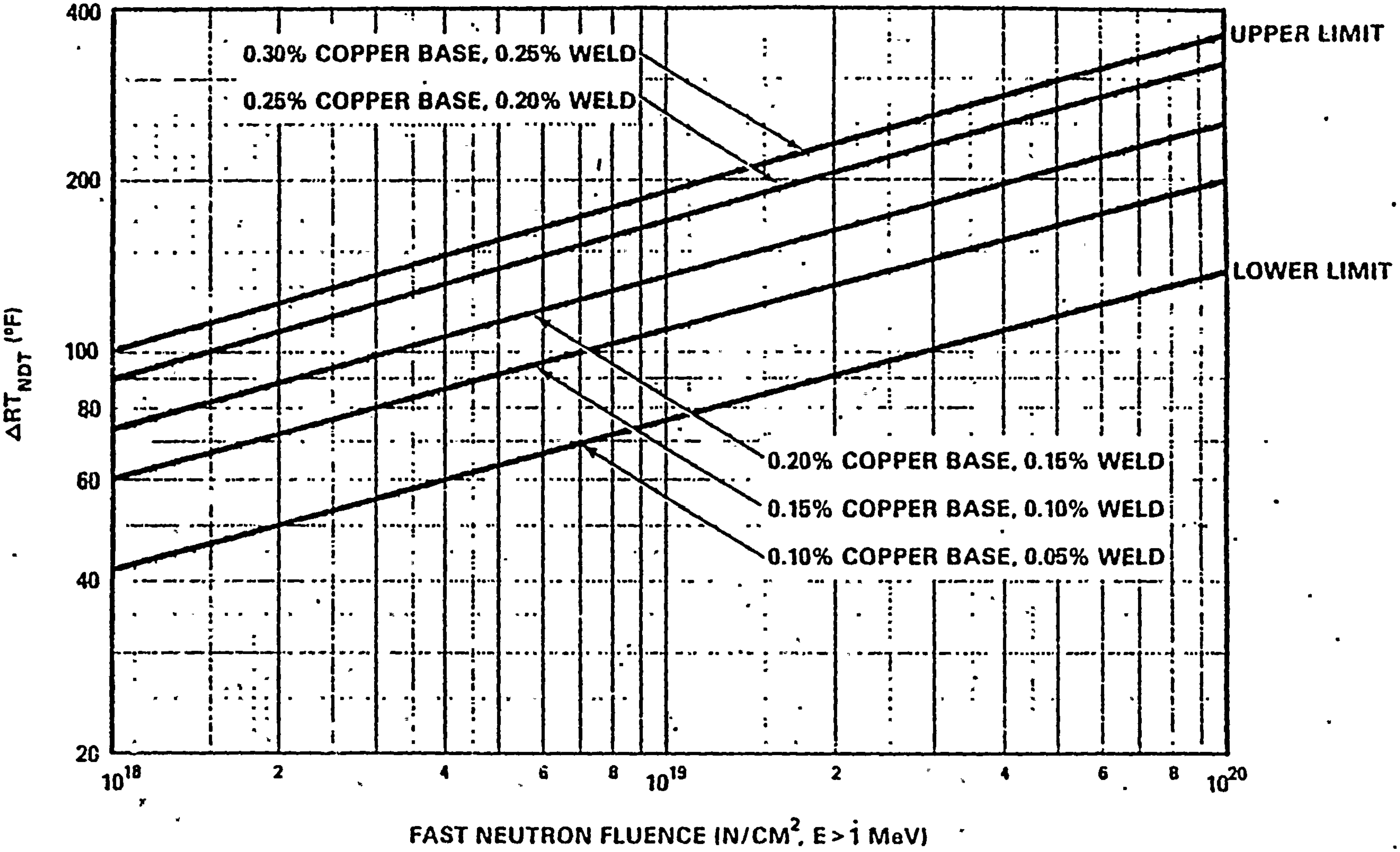

FIGURE B3/4.4-2

EFFECT OF FLUENCE AND COPPER ON SHIFT OF RT_{NDT} FOR REACTOR VESSEL STEELS EXPOSED TO IRRADIATION AT 550°F

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PRESSURE/TEMPERATURE LIMITS (Continued)

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

PRESSURE/TEMPERATURE LIMITS (Continued)

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location

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PRESSURE/TEMPERATURE LIMITS (Continued)

for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

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PRESSURE/TEMPERATURE LIMITS (Continued)

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.45 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to ~~[275]~~°F 250. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a ~~HPSI~~ CHARGING/SAFETY INJECTION pump and its injection into a water-solid RCS.

The Maximum Allowed PORV Setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance of the LTOPS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout ~~of all but one safety injection pump and all~~ but one ~~centrifugal~~ charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The Maximum Allowed PORV Setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

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3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1980 Edition and Addenda through WINTER 1981.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the [reactor vessel head], ~~the [Reactor Coolant System high point],~~ and the [pressurizer steam space], ~~and the [isolation condenser high point]~~ ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

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BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump ~~and one safety injection pump~~ to be OPERABLE and the Surveillance Requirement to ~~verify all~~ ONE charging pump~~s~~ ~~and safety injection pumps except the required~~ OPERABLE ~~charging pump to be inoperable~~ below ~~[275]~~ 250°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 ~~BORON INJECTION SYSTEM [OPTIONAL]~~

The OPERABILITY of the RWST ~~Boron Injection System~~ as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

The limits on RWST ~~injection tank~~ minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. ~~The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.~~

~~[The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensures that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 22,500 ppm boron.]~~

~~3/4.5.5~~ REFUELING WATER STORAGE TANK

~~The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that:~~ (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the

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REFUELING WATER STORAGE TANK (Continued)

RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between [8.5] and [~~11.0~~] for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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BASES

FOR

SECTION 3/4.6A

CONTAINMENT SYSTEMS SPECIFICATIONS

FOR

WESTINGHOUSE

ATMOSPHERIC TYPE CONTAINMENT

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3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to $0.75\ L_a$ or $0.75\ L_t$, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

~~3/4.6.1.4 CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS [OPTIONAL]~~

~~The OPERABILITY of the Isolation Valve and Containment Channel Weld Pressurization Systems is required to meet the restrictions on overall containment leak rate assumed in the safety analyses. The Surveillance Requirements for determining OPERABILITY are consistent with Appendix J of 10 CFR Part 50.~~

3/4.6.1.~~8~~ 4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of ~~[3]~~ -2 psig, and (2) the containment peak pressure does not exceed the design pressure of ~~[54]~~ 45 psig ~~during [LOCA or steam line break conditions].~~

The maximum peak pressure expected to be obtained from a ~~[LOCA or~~ POSTULATED MAIN steam line break~~]~~ event is ~~[45]~~ 39.1 psig. The limit of ~~[3]~~ 1.9 psig for initial positive containment pressure will limit the total pressure to ~~[48]~~ 41 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.~~8~~ 5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a ~~[LOCA or~~ steam line break accident~~]~~. ~~Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.~~

3/4.6.1.~~7~~ 6 CONTAINMENT STRUCTURAL INTEGRITY

[Prestressed concrete containment with ungrouted tendons]

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of [48] psig in the event of a [LOCA or steam line break accident]. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability. (The tendon wire or strand samples will also be subjected to stress cycling tests and to accelerated corrosion tests to simulate the tendon's operating conditions and environment.)

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of proposed Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979, and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

CONTAINMENT STRUCTURAL INTEGRITY (Continued)

~~[Reinforced concrete containment]~~

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of ~~[46]~~ 41 psig in the event of a ~~[LOCA or~~ POSTULATED MAIN steam line break accident]. A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.~~8~~ 7 CONTAINMENT VENTILATION SYSTEM MAKEUP

The [42-inch] containment purge ~~supply~~ and exhaust isolation valves are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a [LOCA or steam line break accident]. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the [8-inch] purge supply and exhaust isolation valves since, unlike the [42-inch] valves, the [8-inch] valves are capable of closing during a [LOCA or steam line break accident]. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during containment PURGING operation. ~~Operation with one pair of these valves open will be limited to [1000] hours during a calendar year. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to support the additional time requests. Only safety-related reasons should be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4 in any calendar year regardless of the allowable hours.~~

Leakage integrity tests with a maximum allowable leakage rate for containment purge ~~supply~~ makeup and exhaust ~~supply~~ valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a [LOCA or steam line break]. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

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CONTAINMENT SPRAY SYSTEM (Continued)

~~[Credit taken for iodine removal]~~

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

~~[No credit taken for iodine removal]~~

~~The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the Containment Spray System, the allowable out-of-service time requirements for the Containment Spray System and Containment Cooling System have been interrelated and adjusted to reflect this additional redundancy in cooling capability.~~

3/4.6.2.2 SPRAY ADDITIVE SYSTEM ~~[OPTIONAL]~~

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. (Later) The limits on NaOH volume and concentration ensure a pH value of between ~~[8.5]~~ and ~~[11.0]~~ for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT FAN COOLERS ~~COOLING SYSTEM~~ ~~[OPTIONAL]~~

The OPERABILITY of the Containment Fan Coolers ~~Containment Cooling System~~ ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

~~[Credit taken for iodine removal by spray systems]~~

The Containment FAN COOLERS ~~Cooling System~~ and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Fan Coolers ~~Cooling System~~ have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

~~[No credit taken for iodine removal by spray systems]~~

~~The Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the Containment Spray System, the allowable out-of-service time requirements for the Containment Cooling System and Containment Spray System have been interrelated and adjusted to reflect this additional redundancy in cooling capacity.~~

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3/4.6.3 IODINE CLEANUP SYSTEM [OPTIONAL]

The OPERABILITY of the containment iodine filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. Operation of the system with the heaters operating for at least 10 hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.6.~~4~~3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.~~5~~4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit ~~[or the Purge System]~~ is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. ~~[Cumulative operation of the Purge System with the heaters operating for 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters].~~ These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

~~The Hydrogen Mixing Systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.~~

~~3/4.6.5~~ PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM [OPTIONAL]

The OPERABILITY of the Penetration Room Exhaust Air Cleanup System ensures that radioactive materials leaking from the containment atmosphere through containment penetrations following a LOCA are filtered and adsorbed prior to reaching the environment. Operation of the system with the heaters operating for at

PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM [OPTIONAL] (Continued)

least 10 continous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the LOCA analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.6.5 VACUUM RELIEF VALVES ~~[OPTIONAL]~~

The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure does not become more negative than -1.93 psig. This condition is necessary to prevent exceeding the containment design limit for internal vacuum of -2 psig.

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3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

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The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% ~~[1100~~ psig~~]~~ of its design pressure of ~~[1000]~~ psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.36×10^7 lbs/h which is 111 % of the total secondary steam flow of 12.2×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For ~~N~~ 3 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

~~For N-1 loop operation~~

$$\cancel{SP = \frac{(X) - (Y)(U)}{X} \times (76)}$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

~~U = Maximum number of inoperable safety valves per operating steam line,~~

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BASES

SAFETY VALVES (Continued)

[109] = Power Range Neutron Flux-High Trip Setpoint for ~~[N]~~ 3 loop operation,

~~[76] = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for [N-1] loop operation,~~

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than [350]°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of ~~[350]~~ 450 gpm at a pressure of ~~[1133]~~ 1170 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of ~~[700]~~ 900 gpm at a pressure of ~~[1133]~~ 1110 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than [350]°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 12 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

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3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of ~~(~~70~~)~~°F and ~~(~~200~~)~~ psig are based on ~~a steam generator RT_{NDT} of ___°F~~ THE AVERAGE IMPACT VALUES OF THE STEAM GENERATOR MATERIAL AT 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

The OPERABILITY of the Emergency Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK ~~[OPTIONAL]~~

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

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ULTIMATE HEAT SINK (Continued)

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 FLOOD PROTECTION [OPTIONAL]

The limitation on flood protection ensures that facility protective actions will be taken (and operation will be terminated) in the event of flood conditions. The limit of elevation ____ Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety-related equipment.

3/4.7.~~7~~ 6 CONTROL ROOM EMERGENCY ~~AIR CLEANUP~~ FILTRATION SYSTEM

The OPERABILITY of the Control Room Emergency ~~Air Cleanup~~ Filtration System ensures that: ~~(1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2)~~ the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.7.~~8~~ 7 ~~ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM~~ REACTOR AUXILARY BUILDING EMERGENCY EXHAUST

The OPERABILITY of the ~~ECCS Pump Room Exhaust Air Cleanup~~ Reactor Auxilary Building Emergency Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

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3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the ~~[Unit Review Group]~~ Manager-Technical Support. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. 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 required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability, one of three functional testing methods is used with the stated acceptance criteria:

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SNUBBERS (Continued)

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, 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.

3/4.7.~~10~~ 9 SEALED SOURCE CONTAMINATION

10CFR31 for By Product Material

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on ~~10 CFR 70.39(a)(3)~~ and limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

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BASES

3/4.7.~~11~~ 10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The Fire Suppression System consists of the water system, ~~spray and/or sprinklers, CO_2, Halon,~~ PREACTION AND MULTICYCLE SPRINKLER SYSTEMS fire hose stations, and yard fire hydrants. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility Fire Protection Program.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met. ~~An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a U.L. or F.M. approved method.~~

In the event the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.~~12~~ 11 FIRE RATED ASSEMBLIES

The functional integrity of the fire rated assemblies and barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. These design features minimize the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishing of the fire. The fire barrier penetrations are a passive element in the facility Fire Protection Program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or (2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established until the barrier is restored to functional status.

3/4.7.13 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ±(LATER)°F.

3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

The OPERABILITY of the Emergency Services Chilled Water System ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

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3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

> INSERT E

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. INSERT F The term, verify, as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

Insert E - Electrical

The SHNPP switchyard is designed using a breaker-and-a-half scheme. The switchyard currently has 5 connections with the CP&L transmission network; each of these transmission lines is considered a physically independent off-site circuit. The SHNPP switchyard has one connection with each of the two Start-up Transformers; the Start-up Transformers are the preferred power source for the Class IE ESF buses. The minimum alignment of off-site power sources will be maintained such that at least two physically independent off-site circuits are available to the switchyard, each Start-up Transformer is energized and the opening of a single circuit breaker in the SHNPP switchyard will not simultaneously interrupt power to both Start-up Transformers. Operation in this configuration provides sufficient redundancy and electrical and physical independence so that no single event is likely to cause simultaneous outage of both circuits which supply power to the on-site class IE ESF distribution systems.

During Modes 5 and 6, the Class 1E buses can be energized from the off-site transmission net work via a combination of the main transformers, and unit auxiliary transformers. This arrangement may be used to satisfy the requirement for one physically independent circuit. ~~during modes 5 and 6.~~

Insert F - Electrical

~~f.~~ Equipment which depend upon the diesel generators for their emergency power but whose total loss would not otherwise require a plant shutdown should not be evaluated in complying with the ~~this~~ Action statement of Section 3.8.1.1 (eg. If Diesel Generator A is out of service and Fuel Handling Building Ventilation System B is out of service, then complying with the ACTION for both trains INOPERABLE is the required ACTION - not a plant shutdown).

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A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers and fuses provide assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

bypassing

The ~~OPERABILITY~~ ~~[or] [bypassing]~~ of the motor-operated valves thermal overload protection ~~[continuously] [or]~~ [during accident conditions] [by integral bypass devices] ensures that the thermal overload protection [during accident conditions] will not prevent safety-related valves from performing their function. [The Surveillance Requirements for demonstrating the ~~[OPERABILITY] [or]~~ [bypassing] of the thermal overload protection ~~[continuously] [and] [or]~~ [during accident conditions] are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.]

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3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS; and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of [2000] ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The ~~locking closed of~~ administrative controls over the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

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3/4.9.6 REFUELING MACHINE ~~MANIPULATOR CRANE~~

REFUELING MACHINE The OPERABILITY requirements for the REFUELING MACHINE ~~manipulator cranes~~ ensure that: (1) ~~manipulator cranes~~ will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING ~~SPENT FUEL STORAGE AREAS~~

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT VENTILATION ~~PURGE AND EXHAUST~~ ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

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3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and ~~STORAGE~~ POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM

3/4.9.12 ~~STORAGE POOL VENTILATION SYSTEM~~

The limitations on the FUEL HANDLING BUILDING EMERGENCY EXHAUST ~~Storage Pool Ventilation~~ System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

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3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to: (1) measure control rod worth, and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which in turn may cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

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3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will 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. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

~~This specification applies to the release of radioactive materials in liquid effluents from all units at the site.~~

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, 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).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operating implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such 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 are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of

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DOSE (Continued)

Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 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.

~~This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. 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 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in 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 dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

~~This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

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3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time 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 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, 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 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

~~This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.~~

The required detection capabilities for radioactive material in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, 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).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established

DOSE-NOBLE GASES (Continued)

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, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, "Revision 1, October 1977 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 are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are 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 10 CFR Part 50, Appendix I," Revision 1, October 1977 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. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM (Continued)

in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of the calculations were: (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.

~~This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. 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 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objectives given in 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 dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

~~This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

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3/4.11.2.5 EXPLOSIVE GAS MIXTURE

GASEOUS RADWASTE TREATMENT

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the ~~WASTE GAS HOLDUP~~ SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of ~~hydrogen and/or~~ oxygen, ~~automatic diversion to recombiners, or injection of dilutants~~ to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4 11.2.6 GAS STORAGE TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981. Since only the gamma body dose factor (DFB_i) is used in the analysis, the Xe-133 equivalent is determined from the DFB_i value for Xe-133 as compared to the composite DFB_i for the actual mixture in the tank.

3/4.11.3 SOLID RADIOACTIVE WASTES

This specification implements the requirements of 10 CFR 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to, waste type, waste pH, waste/liquid/SOLIDIFICATION agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units and from outside

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TOTAL DOSE (Continued)

storage tanks are kept small. 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 40 CFR Part 190 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, with the exception that 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 requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

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3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are 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), Currie, 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).

3/4.12.2 LAND USE CENSUS

This specification 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 given in the ODCM 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 Section IV.B.3 of Appendix I to 10 CFR Part 50. 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.

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3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

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SECTION 5.0

DESIGN FEATURES

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5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area BOUNDARY shall be as shown in Figure ~~[~~5.1-1~~]~~.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure ~~[~~5.1-2~~]~~.

~~MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS~~

~~5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures [5.1-3 and 5.1-4].~~

~~The definition of UNRESTRICTED AREA used in implementing these Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.~~

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment building is a steel-lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 130 feet.

b. Nominal inside height = 160 feet FROM THE LINER ON THE FOUNDATION MAT TO THE SPRING LINE, 225 FEET FROM THE LINER ON THE FOUNDATION MAT TO THE DOME PEAK

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete ~~roof~~ DOME = 2.5 feet.

e. Minimum thickness of concrete floor pad OVER THE CONTAINMENT LINER = 5.0 feet.

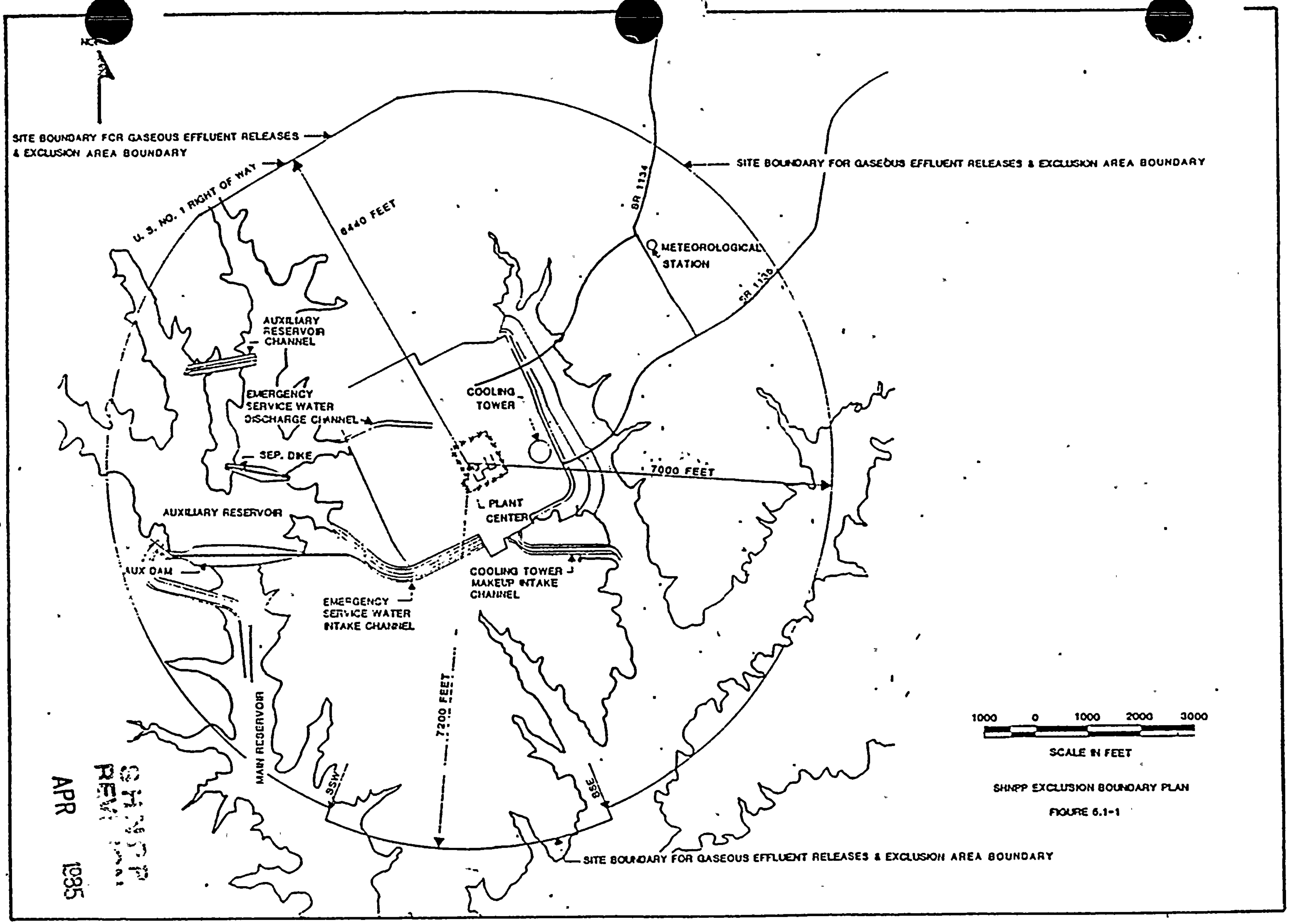
f. Nominal thickness of steel liner = .375 inches IN THE CYLINDRICAL PORTION, 0.25 INCHES ON THE BOTOM AND 0.5 INCHES IN THE DOME.

g. Net free volume = 2.266×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

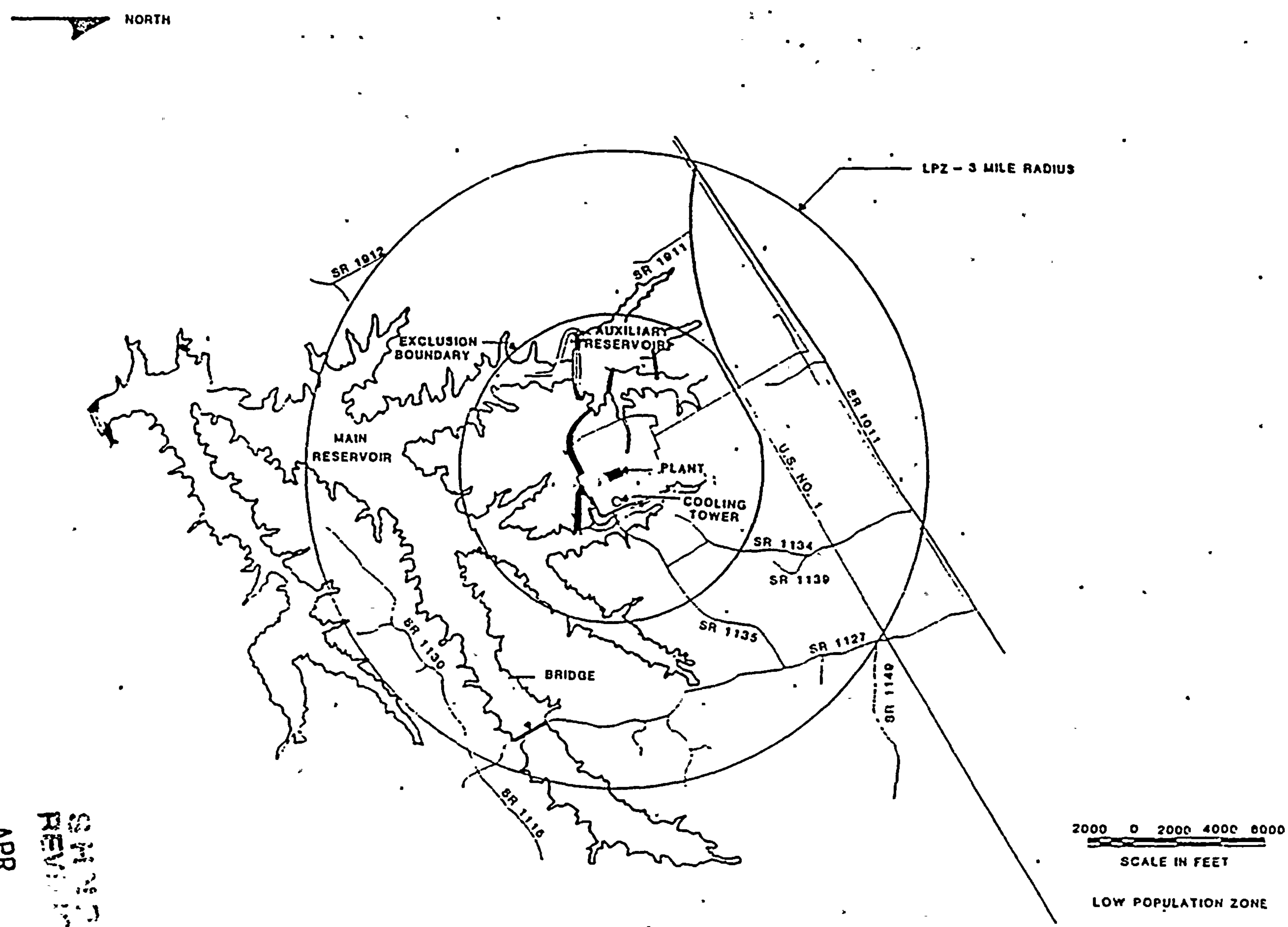
5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a PEAK AIR temperature of 378°F.

5-2



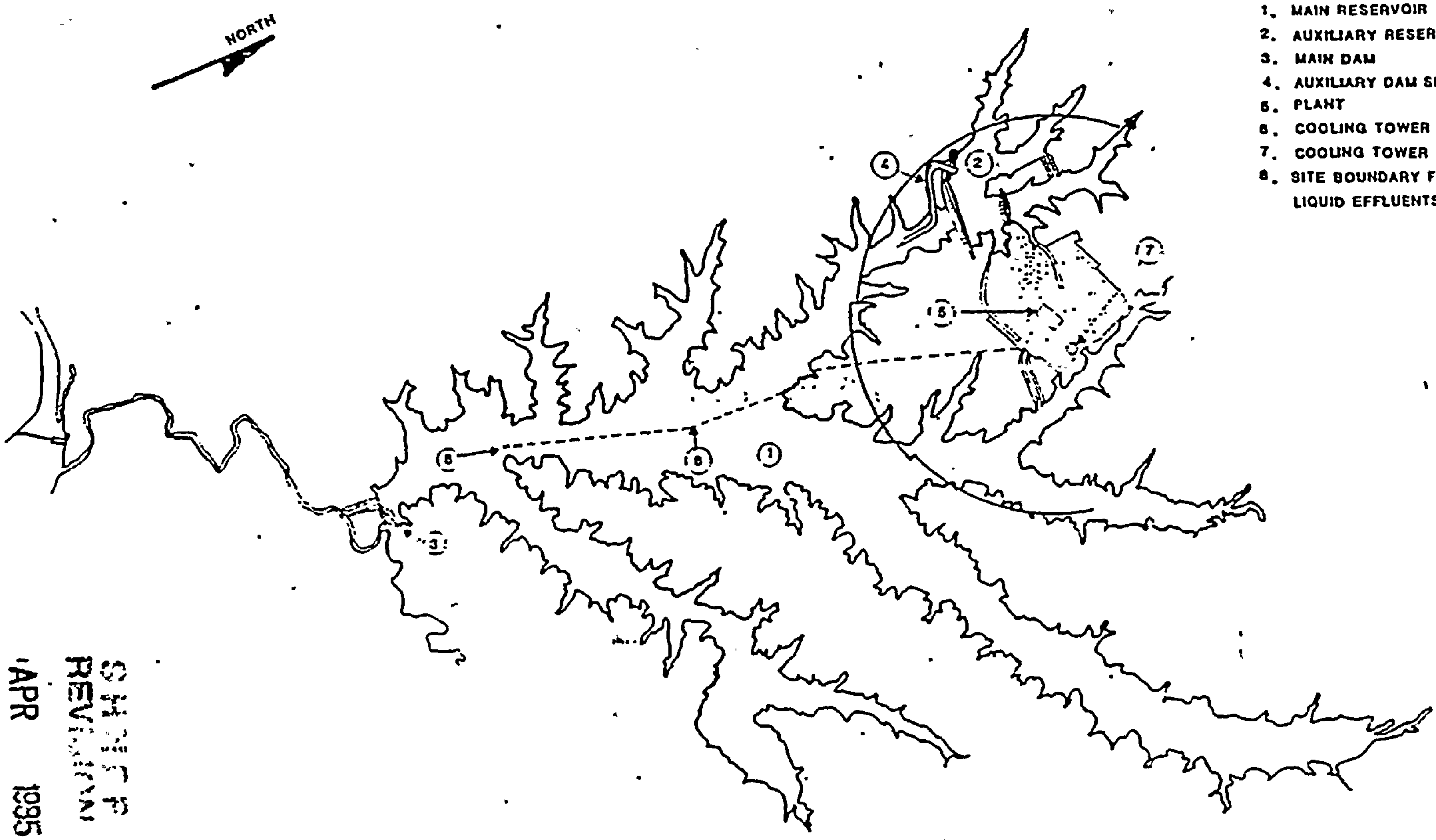
SHNPP EXCLUSION BOUNDARY PLAN

FIGURE 6.1-1



LOW POPULATION ZONE

FIGURE 5.1-2



1. MAIN RESERVOIR
2. AUXILIARY RESERVOIR
3. MAIN DAM
4. AUXILIARY DAM SPILLWAY
5. PLANT
6. COOLING TOWER BLOWDOWN LINE
7. COOLING TOWER
8. SITE BOUNDARY FOR RADIOACTIVE LIQUID EFFLUENTS

0 2000 4000

1000 1000 3000

SCALE IN FEET

SITE BOUNDARY FOR RADIOACTIVE LIQUID EFFLUENTS

FIGURE 5.1-3

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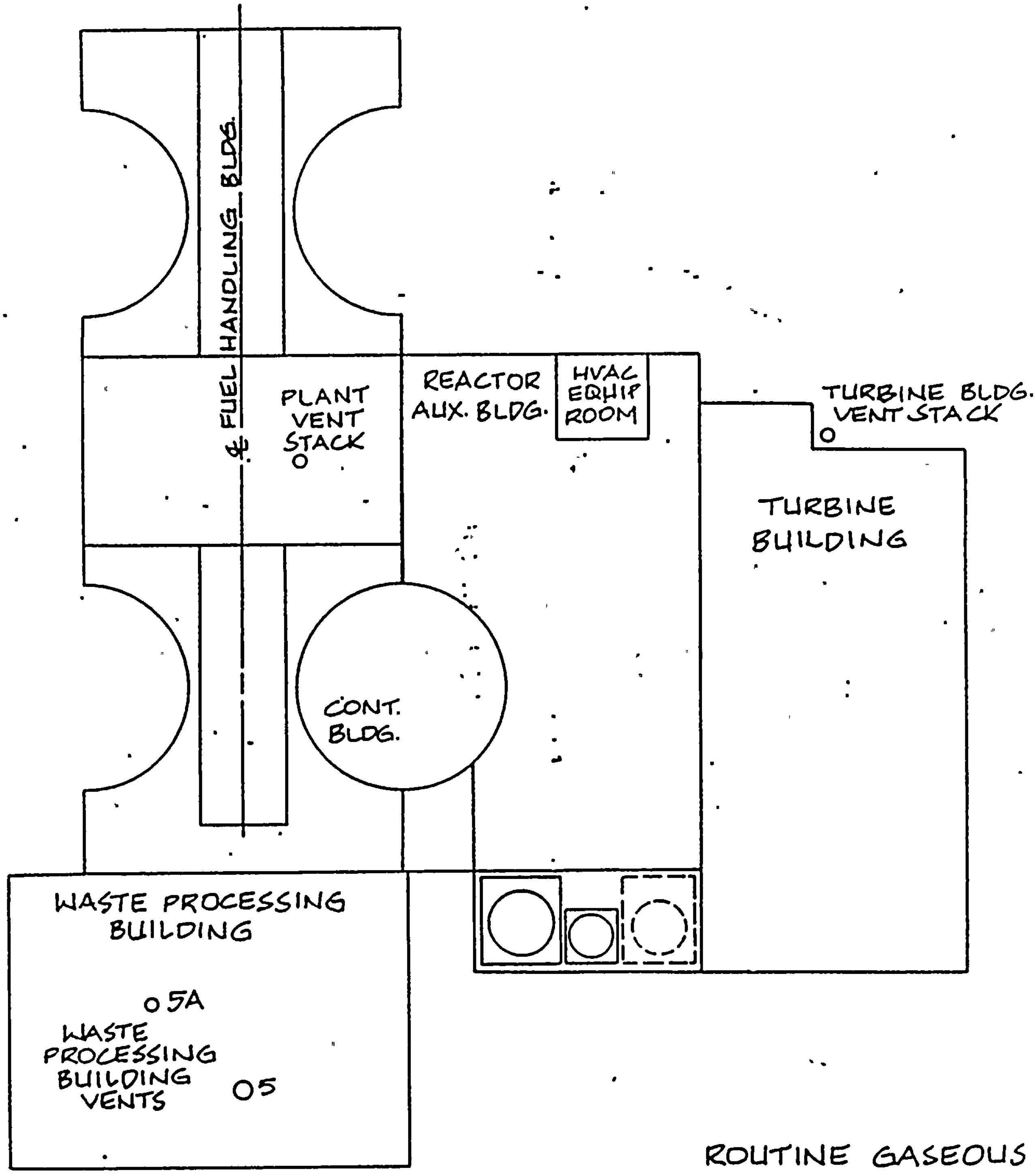

ROUTINE GASEOUS RADIOACTIVE EFFLUENT RELEASE POINTS

FIGURE 5.1-4

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This figure shall consist of a map of the site area and provide at a minimum, the information described in Section [2.1.2] of the FSAR and meteorological tower location.

FIGURE 5.1-1

EXCLUSION AREA

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This figure shall consist of a map of the site area showing the Low Population Zone boundary. Features such as towns, roads, industrial areas and recreational areas shall be indicated in sufficient detail to allow identification of significant shifts in population distribution within the LPZ.

FIGURE 5.1-2

LOW POPULATION ZONE

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This figure shall consist of a map of the site area showing the perimeter of the site and locating the points where gaseous effluents are released. If onsite land areas subject to radioactive materials in gaseous waste are utilized by the public for recreational or other purposes, then these areas shall be identified by occupancy factors and the licensee's method of occupancy control. The figure shall be sufficiently detailed to allow identification of structures and release point locations, and areas within the SITE BOUNDARY that are accessible by members of the general public. See NUREG-0133 for additional guidance.

FIGURE 5.1-3

RESTRICTED AREA AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS EFFLUENTS

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This figure shall consist of a map of the site area showing the perimeter of the site and locating the points where liquid effluents leave the site. If onsite water areas containing radioactive wastes are utilized by the public for recreational or other purposes, the points of release of these water areas shall be identified. The figure shall be sufficiently detailed to allow identification of structures near the release point and areas within the SITE BOUNDARY where ground and surface water is accessible by members of the general public. See NUREG-0133 for additional guidance.

FIGURE 5.1-4

RESTRICTED AREA AND SITE BOUNDARY FOR RADIOACTIVE LIQUID EFFLUENTS

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5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with [Zircaloy-4]. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.9 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 ~~full length~~ control rod assemblies. The ~~full length~~ control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 % silver, 15 % indium, and 5 % cadmium OR 95% HAFNIUM WITH THE REMAINDER ZIRCONIUM. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section [5.2] of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650 °F, except for the pressurizer which is 680 °F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 9410 ± 100 cubic feet at a nominal T_{avg} of ~~[525]~~ 588.8 °F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological ~~tower~~ STATION shall be located as shown on Figure ~~[~~5.1-1~~]~~.

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5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, ~~which includes a conservative~~ AN allowance of ~~[2.6]% Δk/k~~ for uncertainties ~~as~~ IS described in Section ~~[4.3]~~ of the FSAR, and

b. A nominal ~~[21]~~ 10.5 inch center-to-center distance between fuel assemblies placed in the PWR storage racks, AND 6.25 INCH CENTER TO CENTER DISTANCE IN THE BWR STORAGE RACKS.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed ~~[0.98]~~ when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The NEW AND spent fuel storage pools ~~is~~ ARE designed and shall be maintained to prevent inadvertent draining of the pool below elevation 277.5 FEET.

CAPACITY

5.6.3 ~~The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ___ fuel assemblies.~~

THE NEW AND SPENT FUEL STORAGE POOLS ARE DESIGNED FOR A STORAGE CAPACITY OF 1832 PWR FUEL ASSEMBLIES IN FIXED RACKS AND A VARIABLE NUMBER OF PWR AND BWR STORAGE SPACES IN 48 INTERCHANGEABLE 7 x 7 PWR AND 11 x 11 BWR RACKS. THESE INTERCHANGEABLE RACKS WILL BE INSTALLED AS NEEDED. ANY COMBINATION OF PWR AND BWR RACKS MAY BE USED.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

#-STS Shearon Harris Unit 1

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 ~~[250]~~ heatup cycles at ≤ 100°F/h and ~~[250]~~ 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to > 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 ~~[250]~~ pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	200 ~~[100]~~ loss of load cycles, without immediate Turbine or Reactor trip.	> 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 ~~[50]~~ cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 ~~[100]~~ cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 ~~[500]~~ Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 ~~[10]~~ auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	200 ~~[50]~~ leak tests.	Pressurized to ≥ [2485] psig.
	10 ~~[5]~~ hydrostatic pressure tests.	Pressurized to ≥ 3107 ~~[3100]~~ psig.
Secondary Coolant System	[1] steam line break.	Break in a > 6-inch steam line.
	10 ~~[5]~~ hydrostatic pressure tests.	Pressurized to ≥ 1481 ~~[1350]~~ psig.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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6.1 RESPONSIBILITY

6.1.1 The Plant General Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Foreman shall be responsible for unit operations. A management directive to this effect, signed by the Vice President- Harris Nuclear Project, shall be reissued to all plant personnel on an annual basis.

6.2 ORGANIZATION

OFF SITE

6.2.1 The off-site organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The facility organization shall be as shown on Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one Licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in Modes 1, 2, 3, or 4, at least one Licensed Senior Reactor Operator shall be in the Control Room;

c. An individual qualified as a Radiation Control Technician shall be onsite when fuel is in the reactor[1].

d. All CORE ALTERATIONS shall be observed and 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.

e. A Fire Brigade of at least five members shall be maintained on site at all times[1]. The Fire Brigade shall not include the Shift Foreman and the other two members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

---

[1]The Radiation Control Technician and the Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed two hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

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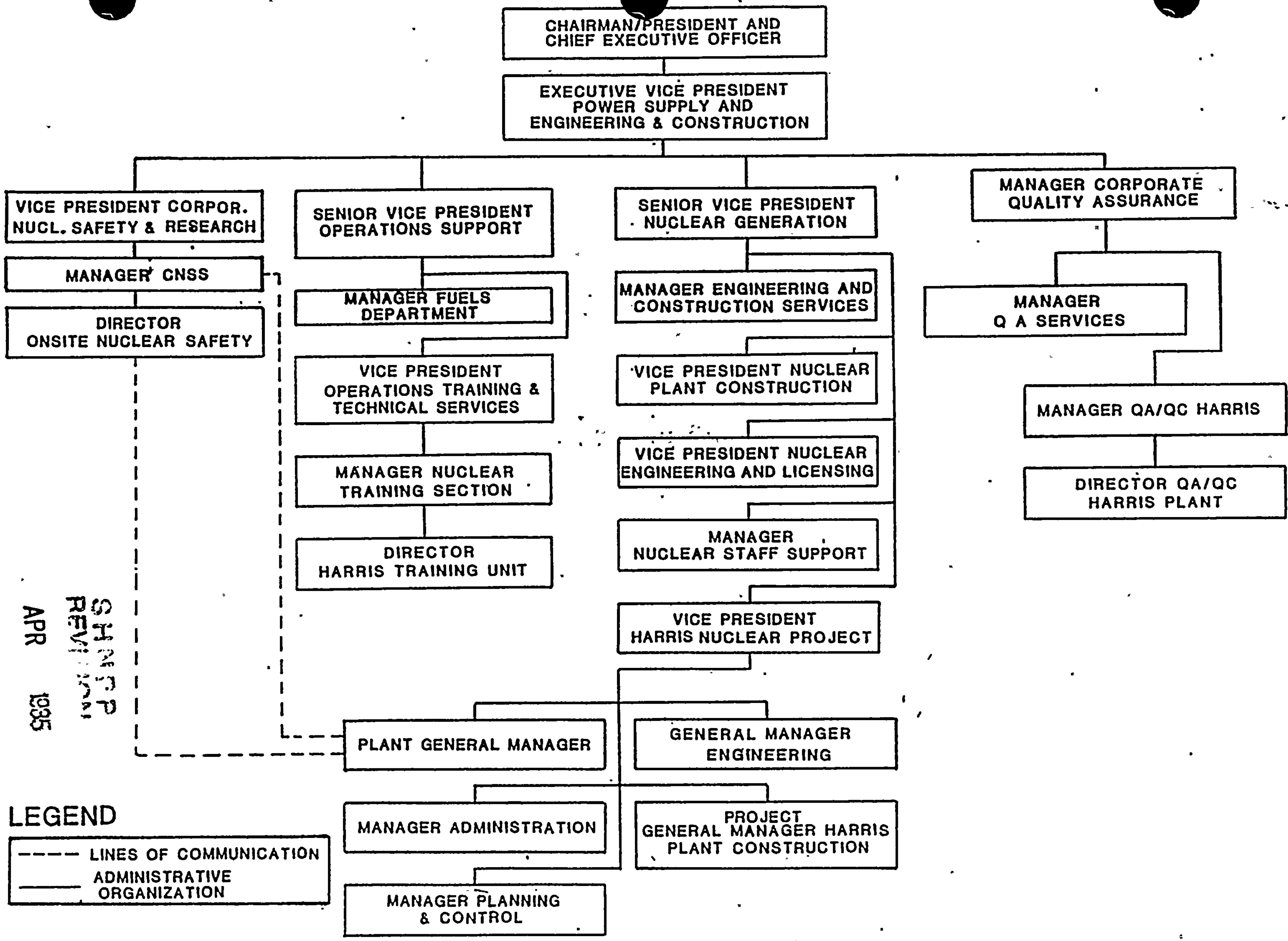
SHEARON HARRIS UNIT 1

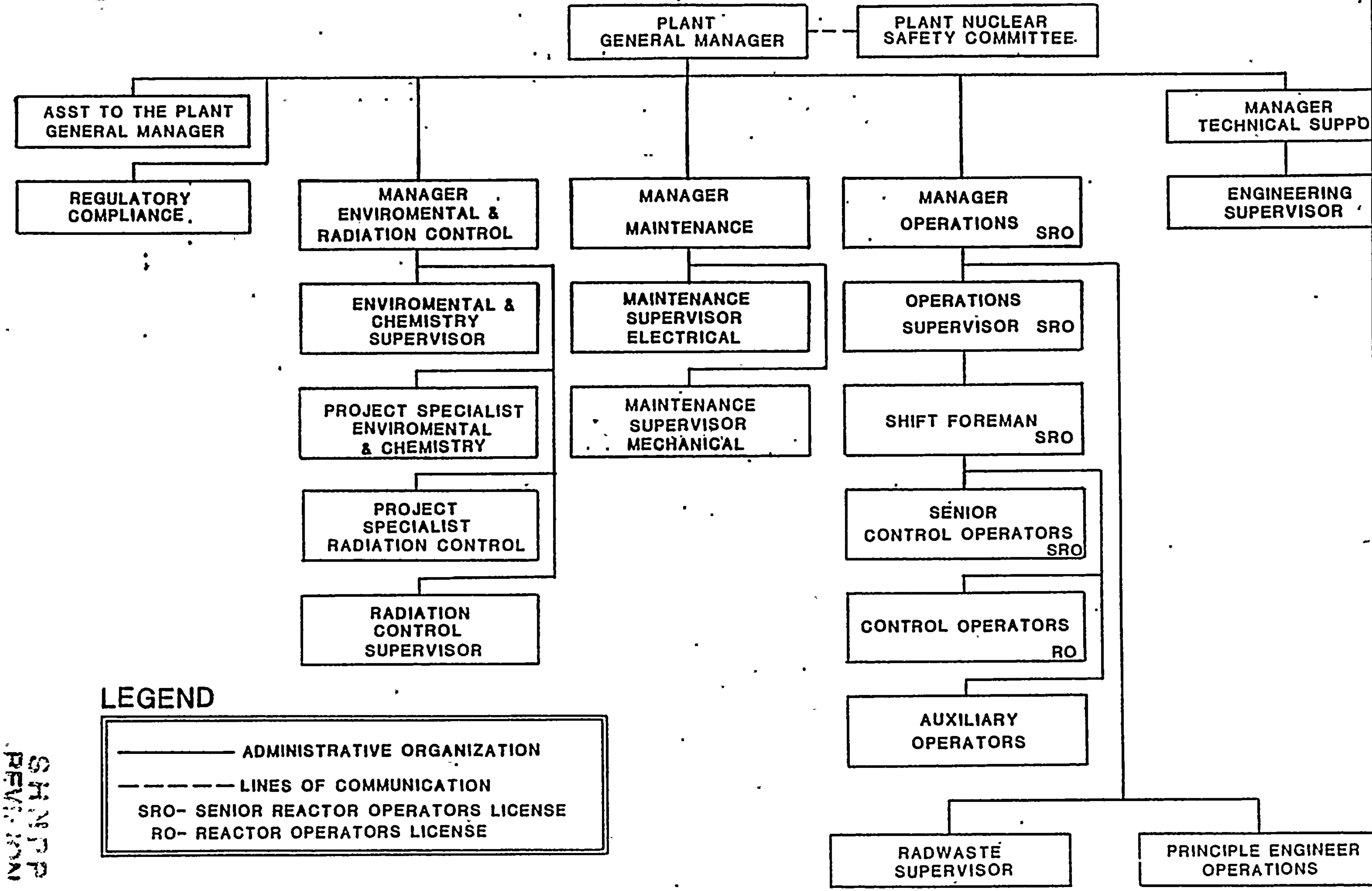
CORPORATE ORGANIZATION



SHEARON HARRIS UNIT 1

FIGURE 6.2-1

PLANT ORGANIZATION



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SHEARON HARRIS UNIT 1

FIGURE 6.2-2

f. Administrative procedures shall be developed and implemented to limit the working hours of facility staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, radiation control technicians, auxiliary operators, and on-shift maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time;

3. STA's are allowed to work a maximum of 84 hours in any seven day period excluding shift turnover time.

4. A break of at least eight hours should be allowed between work periods, including shift turnover time; and

5. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant General Manager, his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant General Manager or delegated to the Manager of the functional area to which the personnel are assigned to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SHEARON HARRIS-UNIT 1

POSITION [a]	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3, & 4	MODES 5 & 6
SF	1	1
SRO	1	None [b]
RO	2	1
Non-Licensed	2	1
STA	1	None

a SF - Shift Foreman with a Senior Reactor Operator's (SRO) License on Unit 1
SRO - Individual with a Senior Reactor Operator's (SRO) License on Unit 1
RO - Individual with a Reactor Operator's License on Unit 1
Non-Licensed - Auxiliary Operator
STA - Shift Technical Advisor

b At least one individual licensed as a licensed SRO or a licensed SRO-Limited to Fuel Handling must be present during CORE ALTERATIONS, this individual shall have no other concurrent responsibilities.

Except for the Shift Foreman, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crew member being late or absent.

During any absence of the Shift Foreman from the Control Room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Foreman from the Control Room while the unit is in MODE 5 or 6, an individual with a valid RO or SRO license shall be designated to assume the Control Room command function.

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6.2.3 ONSITE NUCLEAR SAFETY (ONS)

FUNCTION

6.2.3.1 The ONS Unit shall function to examine facility operating characteristics, NRC issues, industry advisories, and other sources of plant design and operating experience information, which may indicate areas for improving plant safety.

RESPONSIBILITIES

6.2.3.2 The ONS Unit shall be responsible for maintaining surveillance of facility activities which may affect nuclear safety to provide independent verification[2] that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.3 The ONS Unit shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving facility nuclear safety to the Manager - Corporate Nuclear Safety Section.

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall serve in an advisory capacity to the Shift Foreman in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the facility.

6.3 FACILITY STAFF QUALIFICATION

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of the September 1979 draft of ANS 3.1 for comparable position, with the exceptions or alternatives noted in FSAR Section 1.8 - Regulatory Guide 1.8.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Director Harris Training Unit and shall meet or exceed (1) the requirements and recommendations of the September 1979 draft of ANS 3.1 with exceptions or alternatives as noted in FSAR Section 1.8 - Regulatory Guide 1.8, (2) Appendix A of 10CFR Part 55, and (3) the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees and shall include familiarization with relevant industry operational experience identified by the ONS Unit.

---

[2]Not responsible for sign-off function.

6.5 REVIEW AND AUDIT

QUALIFIED SAFETY REVIEWERS

6.5.1 The Plant General Manager shall designate those individuals that will be responsible for performing safety reviews described in Specification 6.5.2. These individuals shall have an academic degree in an engineering or related field or equivalent, and two years of related experience. Such designation shall include the disciplines or procedure categories for which each individual is qualified. Qualified individuals or groups not on the plant staff (as shown on Figure 6.2-2) may be relied upon to perform safety reviews if so designated by the Plant General Manager.

6.5.2 SAFETY REVIEW AND CONTROL

SAFETY EVALUATIONS

6.5.2.1 A safety evaluation shall be prepared for each of the following:

a. Procedures required by Specification 6.8, other procedures that affect nuclear safety, and changes thereto;

b. Proposed tests and experiments that are not described in the Final Safety Analysis Report;

c. Proposed modifications to plant systems or equipment as described in the FSAR;

6.5.2.2 The safety evaluation prepared in accordance with Specifications 6.5.2.1 shall include a written determination, with basis, of whether or not the procedures, or changes thereto; proposed tests and experiments, and changes thereto; and modifications constitute an unreviewed safety question as defined in Paragraph 50.59 of 10 CFR Part 50, or whether they involve a change to the Final Safety Analysis Report, the Technical Specifications, or the Operating License.

6.5.2.3 The safety evaluation shall be prepared by a qualified individual. The safety evaluation shall be reviewed by a second qualified individual.

6.5.2.4 A safety evaluation and subsequent review which conclude that the subject action may involve an unreviewed safety question, a change to the Technical Specifications or a change to the Operating License, will be referred to the Plant Nuclear Safety Committee (PNSC) for their review in accordance with Specification 6.5.3.8. If the PNSC recommendation is that an item is an unreviewed safety question, a change to the Technical Specifications or a change to the Operating License, the action will be referred to the Commission for approval prior to implementation and to the Corporate Nuclear Safety Section for their review in accordance with Specification 6.5.4.9.

6.5.2.5 A safety evaluation and subsequent review which conclude that the subject action does not involve an unreviewed safety question, a change to the Technical Specification, or a change to the Operating License may be approved, as applicable, by the Plant General Manager or his designee, or the Manager of the functional area affected by the procedure, proposed test or experiments, and changes thereto. The individual approving the review shall assure that the reviewers collectively possess the background and qualification in all of the disciplines necessary and important to the specific review.

6.5.2.6 A safety evaluation and subsequent review which conclude that the subject action involves a change in the Final Safety Analysis Report shall be referred to the Corporate Nuclear Safety Section for review in accordance with Specification 6.5.4.9.

6.5.2.7 The individual approving the procedure, test, or experiment or change thereto shall be other than those who prepared the safety evaluation or performed the safety review.

6.5.3 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.3.1 As an effective means for the regular review, overview, evaluation, and maintenance of plant operational safety, a Plant Nuclear Safety Committee (PNSC) shall be established.

6.5.3.2 The PNSC shall function through the utilization of subcommittees, audits, investigations, reports, and/or performance of reviews as a group.

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MEMBERSHIP

6.5.3.3 The PNSC shall be composed of the following:

Chairman - Plant General Manager
Member: Manager - Technical Support
Member: Manager - Operations
Member: Manager - Maintenance
Member: Manager - Environmental and Radiation Control
Member: Assistant to the Plant General Manager
Member: Director - Regulatory Compliance
Member: Director - QA/QC - Harris Plant

ALTERNATES

6.5.3.4 The Chairman may designate in writing other regular members who may serve as Acting Chairman of PNSC meetings. All alternate members shall be appointed in writing by the PNSC Chairman. Alternates shall be designated for specific regular PNSC members and shall have expertise in the same general area as the regular member they represent.

6.5.3.5 All alternates shall, as a minimum, meet equivalent qualification criteria as specified for professional-technical personnel in the September 1979 draft of ANS 3.1.

MEETINGS

6.5.3.6 The Plant Nuclear Safety Committee shall meet at least once per calendar month and as convened by the PNSC Chairman or a designated Acting Chairman. The Plant Nuclear Safety Committee must meet in session to perform its review function. No item involving an unreviewed safety question or a change to the Technical Specifications or the Operating License can be implemented without required PNSC in-session review.

QUORUM

6.5.3.7 The minimum quorum of the PNSC necessary for the performance of activities listed in Specification 6.5.3.8 shall consist of the Chairman or a designated Acting Chairman and three members. No more than two alternates may be counted toward meeting the quorum requirement.

ACTIVITIES

6.5.3.8 The PNSC activities shall include the following:

a. Review of (1) all procedures required by Specification 6.8 and changes thereto and (2) other procedures that affect nuclear safety and changes thereto, any of which (item 1 or 2) have been initially determined to appear to constitute an unreviewed safety question or involve a change to the Technical Specifications;

b. Review of all proposed tests or experiments that have been initially determined to appear to constitute an unreviewed safety question or involve a change to the Technical Specifications;

c. Review of all proposed modifications that have been initially determined to appear to constitute an unreviewed safety question as defined in Paragraph 50.59 of 10CFR Part 50 or involve a change to the Technical Specifications;

d. Review of all proposed changes to the Technical Specifications and Operating License;

e. Review of reports on violations of applicable codes, regulations, orders, Technical Specifications, license requirements, internal procedures, and internal instructions, any of which have nuclear safety significance;

f. Performance of special reviews, investigations (or analyses), and reports thereon as requested by the Plant General Manager or the Manager - Corporate Nuclear Safety Section;

g. Review of all REPORTABLE EVENTS and corrective actions taken to prevent recurrence;

h. Review of facility operations to detect potential nuclear safety hazards;

i. Annual review of the Emergency Plan;

j. Annual review of the Security Plan;

k. Review of unplanned onsite releases of radioactive materials to the environs and corrective actions taken to prevent recurrence of such events; and

l. Review of the changes to the PROCESS CONTROL PROGRAM and OFF-SITE DOSE CALCULATION MANUAL. This review may occur following implementation of the changes; Refer to Specification 6.13 and 6.14.

AUTHORITY

6.5.3.9 The PNSC shall:

a. Render determinations in writing with regard to whether or not each of the items considered under Specification 6.5.3.8.a through 6.5.3.8.c constitute an unreviewed safety question as defined in Paragraph 50.59 of 10CFR Part 50.

b. The PNSC shall provide written notification within 24 hours to the Vice President, Harris Nuclear Project and the Vice President, Corporate Nuclear Safety and Research of disagreement between recommendations of the PNSC and the actions contemplated by the Plant General Manager; however, the course determined by the Plant General Manager to be the most conservative shall be followed.

RECORDS

6.5.3.10 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the provisions of these Technical Specifications. Copies shall be provided to the Vice President - Harris Nuclear Project and the Manager - Corporate Nuclear Safety Section.

6.5.4 CORPORATE NUCLEAR SAFETY SECTION

FUNCTION

6.5.4.1 The Corporate Nuclear Safety Section (CNSS) of the Corporate Nuclear Safety and Research Department shall function to provide independent review of significant plant changes, tests, and procedures; verify that REPORTABLE EVENTS are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer.

ORGANIZATION

6.5.4.2 The individuals assigned responsibility for independent reviews shall be technically qualified in a specified technical discipline or disciplines. These individuals shall collectively have the experience and competence required to review activities in the following areas:

a. Nuclear power plant operations;

b. Nuclear engineering;

c. Chemistry and radiochemistry;

d. Metallurgy;

e. Instrumentation and control;

f. Radiological safety;

g. Mechanical and electrical engineering;

h. Administrative controls;

i. Quality assurance practices;

j. Nondestructive testing; and

k. Other appropriate fields associated with the unique characteristics of the site.

6.5.4.3 The Manager - Corporate Nuclear Safety Section shall have an academic degree in an engineering or related field and, in addition, shall have a minimum of ten years related experience, of which a minimum of five years shall be in the operation and/or design of nuclear power plants.

6.5.4.4 The independent safety review program reviewers shall have an academic degree in an engineering or related field or equivalent and, in addition, shall have a minimum of five years related experience.

6.5.4.5 An individual may possess competence in more than one specialty area. If sufficient expertise is not available within the Corporate Nuclear Safety Section, competent individuals from other Carolina Power & Light Company organizations or outside consultants shall be utilized in performing independent reviews and investigations.

6.5.4.6 At least three individuals, qualified as discussed in Specification 6.5.4.4 above shall review each item submitted under the requirements of Specification 6.5.4.9.

6.5.4.7 Independent safety reviews shall be performed by individuals not directly involved with the activity under review or responsible for the activity under review.

6.5.4.8 The Corporate Nuclear Safety Section independent safety review program shall be conducted in accordance with written, approved procedures.

REVIEW

6.5.4.9 The Corporate Nuclear Safety Section shall perform reviews of the following:

a. All procedures required by Specification 6.8 and other procedures that affect nuclear safety and changes thereto that constitute an unreviewed safety question as defined in Paragraph 50.59 of 10CFR Part 50 or involve a change to the Technical Specifications;

b. All proposed tests or experiments that constitute an unreviewed safety question as defined in Paragraph 50.59 of 10CFR Part 50 or involve a change to the Technical Specifications;

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c. All proposed modifications that constitute an unreviewed safety question as defined in Paragraph 50.59 of 10CFR Part 50 or involve a change to the Technical Specifications;

d. Written safety evaluations for all procedures required by Specification 6.8 and other procedures that affect nuclear safety and changes thereto, and proposed tests or experiments and proposed modifications, any of which constitute a change to the Final Safety Analysis Report. This review may be performed after appropriate management approval; implementation may proceed prior to completion of the review;

e. All proposed changes to the Technical Specifications and Operating License;

f. Violations, deviations, and REPORTABLE EVENTS which require reporting to the NRC such as: violations of applicable codes, regulations, orders, Technical Specifications, license requirements, and internal procedures or instructions having nuclear safety significance, significant operating abnormalities or deviations from normal and expected performance of plant safety-related structures, systems, or components;

g. Reports and minutes of the PNSC; and

h. Any other matter involving safe operation of the nuclear power plant that the Manager - Corporate Nuclear Safety Section deems appropriate for consideration or which is referred to the Manager - Corporate Nuclear Safety Section by the on-site operating organization or other functional organizational units within Carolina Power & Light Company.

6.5.4.10 Review of items considered under 6.5.4.9(g) above shall include the results of any investigations made and the recommendations resulting from these investigations to prevent or reduce the probability of recurrence of the event.

RECORDS

6.5.4.11 Records of Corporate Nuclear Safety Section reviews, including recommendations and concerns, shall be prepared and distributed as indicated below:

a. Copies of documented reviews shall be retained in the CNSS files.

b. Recommendations and concerns shall be submitted to the Plant General Manager and Vice President - Shearon Harris Nuclear Power Plant within 14 days of completion of the review.

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c. A summation of Corporate Nuclear Safety Section recommendations and concerns shall be submitted to the Chairman/ President and Chief Executive Officer; Executive Vice President - Power Supply and Engineering and Construction; Senior Vice President - Nuclear Generation; Vice President - Corporate Nuclear Safety and Research; Vice President - Harris Nuclear Project; Plant General Manager; and other, appropriate, on at least a bimonthly frequency.

6.5.5 <u>CORPORATE QUALITY ASSURANCE AUDIT PROGRAM</u>

<u>FUNCTION</u>

6.5.5.1 The Quality Assurance Services Section of the Corporate Quality Assurance Department shall function to perform audits of facility activities specified in Specification 6.5.5.2.

<u>AUDITS</u>

6.5.5.2 Audits of facility activities shall be performed by the Quality Assurance Services Section. 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 12 months;

b. The training and qualifications of the entire plant staff shown in Figure 6.2-2 at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in plant equipment, structures, systems, or methods of operation that affect nuclear safety at least once per 6 months;

d. The verification of compliance and implementation of the requirements of the Quality Assurance Program to meet the criteria of Appendix B, 10CFR 50, at least once per 24 months;

e. The Emergency Plan and implementing procedures at least once per 12 months;

f. The Security Plan and implementing procedures at least once per 12 months;

g. The Facility Fire Protection Program and implementing procedures at least once per 24 months;

h. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months.

i. The OFF-SITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

j. The PROCESS CONTROL PROGRAM and implementing procedures for SOLIDIFICATION of radioactive wastes at least once per 24 months.

k. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months; and

l. Any other area of facility operation considered appropriate by the Corporate Quality Assurance Services Section.

6.5.5.3 Personnel performing the quality assurance audits shall have access to the plant operating records.

RECORDS

6.5.5.4 Records of audits shall be prepared and retained.

6.5.5.5 Audit reports encompassed by 6.5.5.2 above shall be prepared, approved by the Manager - Quality Assurance Services and forwarded to the Executive Vice President - Power Supply and Engineering and Construction; Senior Vice President - Nuclear Generation; Vice President - Harris Nuclear Project; Vice President - Corporate Nuclear Safety and Research; Plant General Manager; and others, as appropriate, within 30 days after completion of the audit.

AUTHORITY

6.5.5.6 The Manager - Quality Assurance Services Section under the Manager - Corporate Quality Assurance Department shall be responsible for the following:

a. The administering of the Corporate Quality Assurance Audit Program.

b. The approval of the individual(s) selected to conduct quality assurance audits.

6.5.5.7 Audit personnel shall be independent of the area audited.

6.5.5.8 Selection of personnel for auditing assignments shall be based on experience or training that establishes that their qualifications are commensurate with the complexity or special nature of the activities to be audited. In selecting audit personnel, consideration shall be given to special abilities, specialized technical training, prior pertinent experience, personal characteristics, and education.

6.5.5.9 Qualified outside consultants or other individuals independent from those personnel directly involved in plant operation shall be used to augment the audit teams when necessary.

6.5.6 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM

6.5.6.1 An independent fire protection and loss prevention inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

6.5.6.2 An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for a REPORTABLE EVENT:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Paragraph 50.73 to 10CFR Part 50 and Specification 6.9; and

b. Each REPORTABLE EVENT shall be reviewed by the Plant Nuclear Safety Committee and Plant General Manager. The results of this review shall be submitted to the Manager - Corporate Nuclear Safety Section and the Vice President - Harris Nuclear Project. The minutes of the PNSC meeting(s) may be used to document this review.

6.7 SAFETY LIMIT VIOLATION

6.7.1 In addition to the ACTION specified in Specification 2.0, the following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President - Harris Nuclear Project and the Manager - Corporate Nuclear Safety Section shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Nuclear Safety Committee and the Plant General Manager. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the Vice President - Harris Nuclear Project, and the Manager - Corporate Nuclear Safety Section within 14 days of the violation.

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6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978;

b. Refueling operations;

c. Surveillance and test activities of safety-related equipment;

d. Security Plan implementation;

e. Emergency Plan implementation;

f. Fire Protection Program implementation;

g. PROCESS CONTROL PROGRAM implementation;

h. OFFSITE DOSE CALCULATION MANUAL implementation; and

i. Quality Assurance Program for effluent and environmental monitoring.

6.8.2 Each procedure of Specification 6.8.1 shall be reviewed and approved in accordance with Specification 6.5.2.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered;

b. If the change is to be implemented prior to final approval then it shall be approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator License on the affected unit; and

c. The change is documented, reviewed, and approved within 14 days of implementation by the Plant General Manager or by the Manager of the functional area affected by the procedure, if already implemented.

6.8.4 The following programs shall be established, implemented, and maintained:

a. <u>Reactor Coolant Sources Outside Containment</u>

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A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include RHR, Safety Injection, Chemical and Volume Control, Containment Spray, Post-Accident Sample System, and Post-Accident RAB Ventilation System and Valve Leakoff Equipment Drain System as specified in FSAR Section TMI-III.D.1.1. The program shall include:

1) Preventive maintenance in accordance with licensee approved procedures;

2) Periodic visual inspection; and

3) Integrated leak testing for each system at refueling cycle intervals or less.

b. <u>In-Plant Radiation Monitoring</u>

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel;

2) Procedure for monitoring; and

3) Preventive maintenance of sampling and analysis equipment in accordance with licensee approved procedures.

c. <u>Secondary Water Chemistry</u>

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables;

2) Identification of the procedures used to measure the values of the critical variables;

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage;

4) Procedures for the recording and management of data;

5) Procedures defining corrective actions for all off-control point chemistry conditions; and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1) Training of personnel; and

2) Procedures for monitoring.

e. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines, and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel;

2) Procedures for sampling and analysis; and

3) Preventive maintenance of sampling and analysis equipment in accordance with licensee approved procedures.

f. Inspections of Water Control Structures

A program to implement an ongoing inspection program in accordance with Regulatory Guide 1.127 (Revision 1, March 1978) for the Main and Auxiliary Dams, the Auxiliary Separating Dike, the Emergency Service Water and Discharge Channels, and the Auxiliary Reservoir Channel. The program shall include the following:

1. The provisions of Reg. Guide 1.127, Revision 1, to be implemented as a part of plant start-up operations.

2. Subsequent inspections at yearly intervals for at least the next three years. If adverse conditions are not revealed by these inspections, inspection at five year intervals will be performed.

3. The program shall specify a maximum sediment depth that will be permitted to accumulate in the channels before removal is required.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE EVENTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator, United States Nuclear Regulatory Commission, Region II, unless otherwise noted.

START-UP REPORTS

6.9.1.1 A summary report of plant start-up and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the Operating License involving a planned increase in power level, (3) installation of fuel that has a diffe rent 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.

6.9.1.2 The Start-up Report shall address each of the tests identified in the FSAR and shall 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.

6.9.1.3 Start-up reports shall be submitted within (1) 90 days following completion of the start-up test program, (2) 90 days following resumption or commencement of commercial power operation, (3) 9 months following initial criticality, whichever is earliest. If the Start-up Report does not cover all three events; i.e., initial criticality, completion of start-up 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.

ANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the unit as described below during the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality. The annual report shall document all challenges to the Pressurizer PORV's and Safety Relief Valves.

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6.9.1.5 Reports required on an annual basis shall include 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 [3], e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated, based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources shall be assigned to specific major work functions.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine Annual Radiological Annual Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with 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 land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include 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 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 1, November 1979. In the event that some individual results are not available for inclusion with the report, 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.

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[3]This tabulation supplements the requirements of Paragraph 20.407 of 10 CFR Part 20.

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The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps[4], covering all sampling locations keyed to a table giving distances and directions from the center line of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, and the corrective action taken if the specified program is not being performed required by Specification 3.12.3; reason for not conducting the Radiological Environmental Monitoring Program as required by Specification 3.12.1 and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample mearsurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to Action b of specification 3.12.1 and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.7 Routine Semiannual Radioactive Effluent Release reports covering the operation of the unit during the previous six months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste release from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid waste (as defined by 10 CFR Part 60), type of container (e.g., LSA, Type A, Type B, Large Quantity), and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release report to be submitted within 60 days after January 1 of each year shall include:

a. An annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability[5].

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[4]One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

[5]In lieu of submission with the Semiannual radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data in a file that shall be provided to the NRC upon request.

b. An assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year.

c. An assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the EXCLUSION AREA BOUNDARY (Figures 5.1-1 and 5.1-3) during the reporting period.

All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. Historical annual average meteorology or meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFF-SITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PCP and to the ODCM, pursuant to Specification 6.13 and 6.14 respectively; a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2; and any major changes to Liquid, Gaseous or Solid Radwaste Treatment Systems pursuant to 6.15.

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The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specifications 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specifications 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Region II Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

RADIAL PEAKING FACTOR LEVEL REPORT

6.9.1.9 The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the NRC Regional Office, with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing Bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it shall be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region II within the time period specified for each report.

6.10 RECORD RETENTION

Facility records shall be retained in accordance with ANSI-N45.2.9-1974 as described in FSAR Section 1.8,-Regulatory Guide 1.88.

6.10.1 The following records shall be retained for at least five years:

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a. Records and logs of facility operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS submitted to the Commission;

d. Records of the performance of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burn up histories;

c. Records of facility radiation and contamination surveys.

d. Records or radiation exposure for all individuals entering radiation control areas;

e. Records of gaseous and liquid radioactive material released to the environs;

f. Records of transient or operational cycles for those facility components identified in Table 5.7-1;

g. Records of reactor tests and experiments;

h. Records of training and qualification for current members of the plant staff (refer to Figure 6.2-2);

i. Records of in-service inspections performed pursuant to these Technical Specifications;

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j. Records of Quality Assurance activities required by the QA Program;

k. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to Paragraph 50.59 of 10CFR Part 50;

l. Records of (1) meetings of the PNSC, and (2) the independent reviews performed by the Corporate Nuclear Safety Section;

m. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records;

n. Records of secondary water sampling and water quality; and

o. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 <u>RADIATION PROTECTION PROGRAM</u>

6.11.1 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 <u>HIGH RADIATION AREA</u>

6.12.1 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 (RWP)[6],. 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; or

[6]Radiation Control personnel or personnel escorted by Radiation Control personnel may be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

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b. A radiation monitoring device which continuously integrates the radiation does 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; or

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 facility Radiation Control Supervisor or his designee in the Radiation Work Permit.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or Radiation Control supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 <u>PROCESS CONTROL PROGRAM (PCP)</u>

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

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2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the Manager - Operations.

b. Shall become effective upon review and acceptance by the Manager - Operations.

6.14 <u>OFFSITE DOSE CALCULATION MANUAL (ODCM)</u>

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM changed with each page numbered, dated and containing the revision number together with appropriate analyses of evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or Setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the ~~Plant General~~ Manager-E&RC.

b. Shall become effective upon review and acceptance by the Manager-E&RC

6.15 <u>MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS</u>[7]

6.15.1 Licensee-initiated major changes to the Radioactive Waste Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with Paragraph 50.59 of 10CFR Part 50;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable in accordance ~~to~~ WITH appropriate plant modification procedures.

b. Shall become effective upon review and acceptance.

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[7]The Licensee may choose to submit the information called for in this specification as part of the annual FSAR update.

APPENDIX B

TO FACILITY OPERATING LICENSE NO.________.
SHEARON HARRIS NUCLEAR POWER PLANT
UNIT NO. 1

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-400

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

(NON-RADIOLOGICAL)

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1.0 DEFINITIONS

Annually: As defined in the NPDES Permit.

Biweekly: As defined in the NPDES Permit.

CP&L: Carolina Power & Light Company

Daily Average: As defined in the NPDES Permit.

Daily Maximum: As defined in the NPDES Permit.

EIS: Environmental Impact Statement.

ES: Environmental Specifications.

FES-OL: Final Environmental Statement - Operating License.

Instrument Maximum As defined in the NPDES Permit.

Maximum Roving Average: As defined in the NPDES Permit.

Monthly: As defined in the NPDES Permit.

Normal Operation: Operation of any unit at the plant at greater than 5 percent of rated thermal power in other than a safety or power emergency situation.

NPDES Permit: NPDES permit is the current National Pollutant Discharge Elimination System Permit issued by United States Environmental Protection Agency or the North Carolina Department of Natural Resources and Community Development (NCDNRCD) to Carolina Power & Light Company as pertains to Shearon Harris Nuclear Power Plant (SHNPP) Unit 1. This permit authorizes CP&L to discharge controlled waste waters from the SHNPP into the waters of the State of North Carolina.

Site: On-site includes the area within the exclusion area boundary and the area encompassed by the 243.0 ft. contour of the Main Reservoir and the 260.0 ft. contour of the Auxiliary Reservoir as specifically described in FSAR Section 2.1.1. Off-site includes all other areas.

Plant: Plant refers to SHNPP Unit 1.

Twice Yearly: As defined in the NPDES Permit.

NEPA: National Environmental Policy Act.

USEPA: United States Environmental Protection Agency, an agency of the United States Government.

NRC: Nuclear Regulatory Commission.

Weekly: As defined in the NPDES Permit.

2.0 Limiting Conditions for Operation

2.1 Non-radiological Limits

Not Applicable

3.0 Environmental Monitoring

In compliance with the provisions of the Clean Water Act (33 USC Section 1251, et seq.) and in the interest of avoiding duplication of effort, the conditions and monitoring requirements related to water quality and aquatic biota are specified in the National Pollutant Discharge Elimination System (NPDES) Permit issued by the U. S. Environmental Protection Agency and/or North Carolina DNRCD to Carolina Power & Light Company. This permit authorizes CP&L to discharge controlled waste water from the SHNPP into specified waters of the State of North Carolina.

3.1 Nonradiological Monitoring

The Nuclear Regulatory Commission will be relying on the NPDES permit for protection of the aquatic environment from non-radiological effluents.

4.0 Special Studies and Requirements

4.1 Exceptional Occurrences

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4.1.1 Unusual or Important Environmental Events

Requirements

The licensee shall record any occurrence of unusual or important events which are observed by management or other qualified personnel. In conjunction with any required monitoring program, the licensee shall document an occurrence of unusual or important events that could indicate potential environmental impact causally related with station operation. The following are examples: significant on-site flora or fauna disease outbreaks; unusual mortality of any species protected by the Endangered Species Act of 1973; significant fish kills according to the definition of the State of North Carolina near or downstream of the site.

This special requirement shall commence with the date of issuance of these environmental Technical Specifications and continue until approval for modification or termination is obtained from the NRC in accordance with Subsection 5.6.1.

Action

Copies of the biological monitoring reports filed with NCDNRCD shall be concurrently submitted to NRC.

Bases

Providing reports to the NRC of extraordinary or significant events as described above is necessary for responsible and orderly regulation of the nation's system of nuclear power reactors. Notification to NRC may serve to alleviate the magnitude of the environmental impact or to place it into a perspective broader than that available to the licensee. The information thus provided may be useful or necessary to others concerned with the same environmental resources. NRC also has an obligation to be responsive to inquiries from the public and the news media concerning potentially significant environmental events at nuclear power plants.

4.1.2 Exceeding Limits of Other Relevant Permits

Requirements

The licensee shall notify the NRC of occurrences exceeding the limits specified in relevant permits and certificates issued by other federal, state and local agencies by providing to the NRC a copy of the notice as submitted to the relevant agency.

This special requirement shall commence with the date of issuance of these environmental specifications contained herein and continue until approval for modification or termination is obtained from the NRC in accordance with Subsection 5.6.

Action

The licensee shall provide the NRC copies of reports to NPDES cognizant agencies in the event of excursion beyond a limit specified in a permit or certificate issued by another federal, state or local agency.

Bases

NRC is required under NEPA to maintain an awareness of environmental impacts causally related with the construction and operations of facilities licensed under its authority.

4.2 Biological Monitoring Program

Requirements

The licensee shall provide the results of biological studies when the results of such studies are required by the NPDES permit issuing agency.

Action

The licensee shall submit informational copies of biological studies in accordance with the schedule required by the NPDES Permit.

Bases

The preoperational non-radiological (biological) monitoring program required in the Revised Final Environmental Statement will be conducted until one year after the unit is in commercial operation. Future monitoring programs beyond that described above will be governed by the NPDES permit.

The submittal of results from the programs required by the NPDES Permit will allow the staff to follow the consequences of the NRC licensing action.

5.0 Administrative Controls

5.1 Responsibility

The Plant General Manager has the responsibility for operating the plant in compliance with these Specifications. Management responsibilities for the biological monitoring programs referenced in the Environmental Specifications rests with the Manager of the Environmental Services Section who reports to the Senior Vice President, Operations Support.

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5.2 Review and Audit

5.2.1 Independent Review

Independent review and audit of plant operations and specifications for environmental matters will be performed by Corporate Quality Assurance Department. The Corporate Quality Assurance Department reports to the Executive Vice President.

5.3 Procedures

5.3.1 Normal Operating Procedures

Written procedures shall be prepared and followed to implement the Environmental Specifications. They shall be subject to audit. These procedures will be reviewed and approved by appropriate supervisors.

5.3.4 Changes in Practices, Plant Design or Operation

Changes in practices, plant design or operation may be made subject to conditions described below:

a) The licensee may (1) make changes in the plant design and operation, (2) make changes in the environmental programs described in the NPDES Permit and (3) conduct tests and experiments not described in the NPDES Permit without prior Commission approval, unless the proposed change, test or experiment involves an unreviewed environmental question as defined in b below.

b) A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the final environmental impact statement as modified by staff's testimony to the Atomic Safety and Licensing Board, supplements thereto, environmental impact appraisals, or in initial or final adjudicatory decisions; or (2) a significant change in effluents or power level as specified in 51.5(b) of 10 CFR 51; or (3) a matter not previously reviewed and evaluated in the documents specified in (1) of this section which may have a significant adverse environmental impact. The Plant General Manager shall establish procedures to decide if a proposed change, test or experiment constitutes an unreviewed environmental question.

c) The licensee shall maintain records of changes in procedures and in facility design or operation made pursuant to these specifications. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph "A" of this subsection.

d) Changes in the NPDES shall be governed by NCDNRCD.

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5.4 Plant Reporting Requirements

Reports will be made as required in sections 4.1.1, 4.1.2 and 4.2.

5.5 Changes in Environmental Specifications and Permits

Changes and additions to required Federal (other than NRC), State, local and regional authority permits and certificates for the protection of the environment that pertain to the requirements of these Environmental Specification shall be reported to the NRC. In the event that the licensee initiates or becomes aware of a request for changes to any of the water quality requirements, limits or values stipulated in any certification or permit issued pursuant to Section 401 or 402 of the Clean Water Act which is also the subject of an Environmental Specifications reporting requirement, NRC shall be notified.

If a permit or certification, in part or in its entirety, is appealed and stayed, and if this causes water quality requirements of Sections 401 or 402 of the Clean Water Act to become nonapplicable, NRC shall be notified as described above. If, as a result of the appeal process, the 401 and 402 requirements are changed, the change shall be dealt with as described in the previous paragraph of this section.

5.6 The following records shall be retained for three years.

a) Records of changes to the Environmental Program including, when applicable, records of NRC approval of such changes.

b) Records of modifications to plant structures, systems and components determined to potentially affect the continued protection of the environment.

c) Records of changes to permits and certifications required by federal (other than NRC), state, local and regional authorities for the protection of the environment.

d) Routine reports submitted to the NRC.

e) Records of review and audit activities.

f) Events, and the reports thereon, which are the subject of nonroutine reports to the NRC.

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APPENDIX B

Attachment 2

DEVELOPMENT OF TECHNICAL SPECIFICATIONS FOR Harris 1