DCS

**Colt Industries**

**Fairbanks Morse**
**Engine Division**
701 Lawton Avenue
Beloit, Wisconsin 53511
608/364-4411

February 4, 1986

Office of Inspection & Enforcement
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Attention: Mr. James G. Keppler
Director, Region III

Subject: Colt Industries-Fairbanks Morse Engine Division
Colt PC-2 Series Emergency Diesel Generators
Potential Overspeed Trip
Part 21 Report

Enclosure: 1) Colt's Diagram of Rack Boost
Cylinder Control Connections

Gentlemen:

An overspeed trip on startup has been reported to Colt by Public Service of New Hampshire, Seabrook Nuclear Plant, Unit 1B. Reports from the site indicate the starting air pressure, which actuates the governor rack boost, was not being vented from the rack boost promptly after the start signal had been terminated, causing the boost to fight the governor. Examination of the piping schematics confirmed this could happen, an alternate piping arrangement was suggested, and a field change was initiated. A Colt engineer was dispatched to observe tests confirming the validity of the change.

Site tests and engineering evaluation completed this date have confirmed the repiping as shown on the enclosure removes the potential problem. The potential problem is limited to PC-2 Series of engines (this report does not apply to 38TD8-1/8 engines).

Utilities that have PC-2 series engines are:

- Alabama Power Co., Farley Units 1 & 2. 348, 364
- Public Service of Indiana, Marble Hill.
- Public Service of New Hampshire, Seabrook 1 & 2.
- Public Service Electric & Gas Co., Hope Creek.
- Northeast Utilities, Millstone III.
- South Carolina Electric & Gas Co., Summer Station.
- Duquesne Light Co., Beaver Valley Unit 2.
- Long Island Lighting Co., Shoreham Station.
- Kansas Gas & Electric, Wolf Creek Station
- Union Electric, Callaway Station
- Washington Public Power Supply System, Units 3 & 5.





IE19

A copy of this report and enclosure 1 is being forwarded to the Utilities involved by registered mail. The modification involves only minor tubing changes which should be implemented by each utility at their sites.

Sincerely,

J. M. Moriarty
Manager
Utility Sales

JMM
ems

cc: U.S. Nuclear Regulatory Commission
% Document Management Branch
Washington, D.C. 20555

Stone & Webster Engineering Corp.
P.O. Box 2325
Boston, MA 02107
Attn: Lead Electrical Engr.
J.O.#12179(Millstone III)

So. Carolina Electric & Gas Co.
P.O. Box 764
Columbia, S.C. 29218
Attn: Mr. Dan Nauman
Mgr., Q.A. & Security

Stone & Webster Engineering Corp.
P.O. Box 2325
Boston, MA 02107
Attn: Project Eng. for
Duquesne J.O. 12241

Stone & Webster Engineering Corp.
P.O. Box 2325
Boston, MA 02107
Attn: Proj. Engineer LILCO

Washington Public Power SUpply System
P.O. Box 1189
Elma, WA 98541
Attn: Mr. Tom Murawski

Bechtel Power Corp.
P.O. Box 3965
San Francisco, CA 94119
Attn: Hope Creek Project Engr.

United Engineers & Constructors,Inc.
P.O. Box 8223
Philadelphia, PA 19101
Attn: Mr. D.H. Rhoads
Proj.Engrg.Mgr. (Seabrook)

Sargent & Lundy Engineers
55 E. Monroe St.
Chicago, IL 60603
Attn: Mr. P.L. Wattelet
Marble Hill Proj. Mgr.

Alabama Power Company
P.O. Drawer 470
Ashford, AL 36312
Attn: Mr. George Hairston

Bechtel Power Corp.
P.O. Box 607
Gaithersburg, MD 20760
Attn: SNUPPS Proj. Engineers

## RACK BOOST CYLINDER CONNECTIONS - COLT PIELSTICK ENGINES IN NUCLEAR SERVICE

### The Problem:

On one of the engines at the Seabrook Nuclear Power Plant of Public Service of New Hampshire, there has been two occurrences of the engine tripping out on overspeed on being started for test purposes. Both of the recorded occurrences seem to have involved the rack boost cylinder on the control system. The purpose of the rack boost cylinder is to ensure that the engine receives fuel (that the injection pump racks come off of 'zero') in the early stages of engine starting (upon the engine initially rotating) to ensure fast starting capability. This is desired because the governor is not capable of moving the racks until sufficient pressure builds up in the governor actuator after the engine begins to rotate at some minimum speed. Once the governor actuator builds up operating pressures, the governor then further increases the rack setting until the engine attains rated speed (the speed for which the governing system is set). The rack boost cylinder is only required at the very early state of the starting process, and in fact, the governor must push the rack boost cylinder back to the 'no fuel' position once the engine is up to rated speed.

Investigation of the problem at Seabrook Station revealed that the source of air pressure for the rack boost cylinder was staying on (at a high pressure) for up to 15 seconds beyond the time that the air start process was otherwise terminated. This is due to the fact that the main air start headers, to which the rack boost cylinder is connected through a series of shuttle valves, are not specifically vented at the end of the starting air admission period. It was always thought that the air pressure in these headers was vented out through the engine, and/or the air start distributor assembly. It appears that this is not necessarily the case.

To prevent any further occurences, or the possibility of same, Colt Industries is recommending that all Colt-Pielstick engines in nuclear service be modified to positively vent the air from the rack boost cylinder, and its associated shuttle valves, by making the source of that air supply from the pilot air lines that control the main air start admission valves. The pilot air lines are controlled by solenoid valves that receive the start signal from the control system. These are three way valves which positively vent the pilot signal to atmosphere when the control system senses that the engine is started.

Testing conducted at Seabrook Station on the unit that had experienced the problem demonstrated the positive action of this change in air source for the rack boost cylinder.

### The Modification:

The modification of the system consists of removing the rack boost shuttle valve connections to the main air start headers, and making those connections to the pilot air lines, at the point where the pilot air lines connects to the pilot connection on the main air start valves. This reconnection is shown schematically on the enclosed figure.

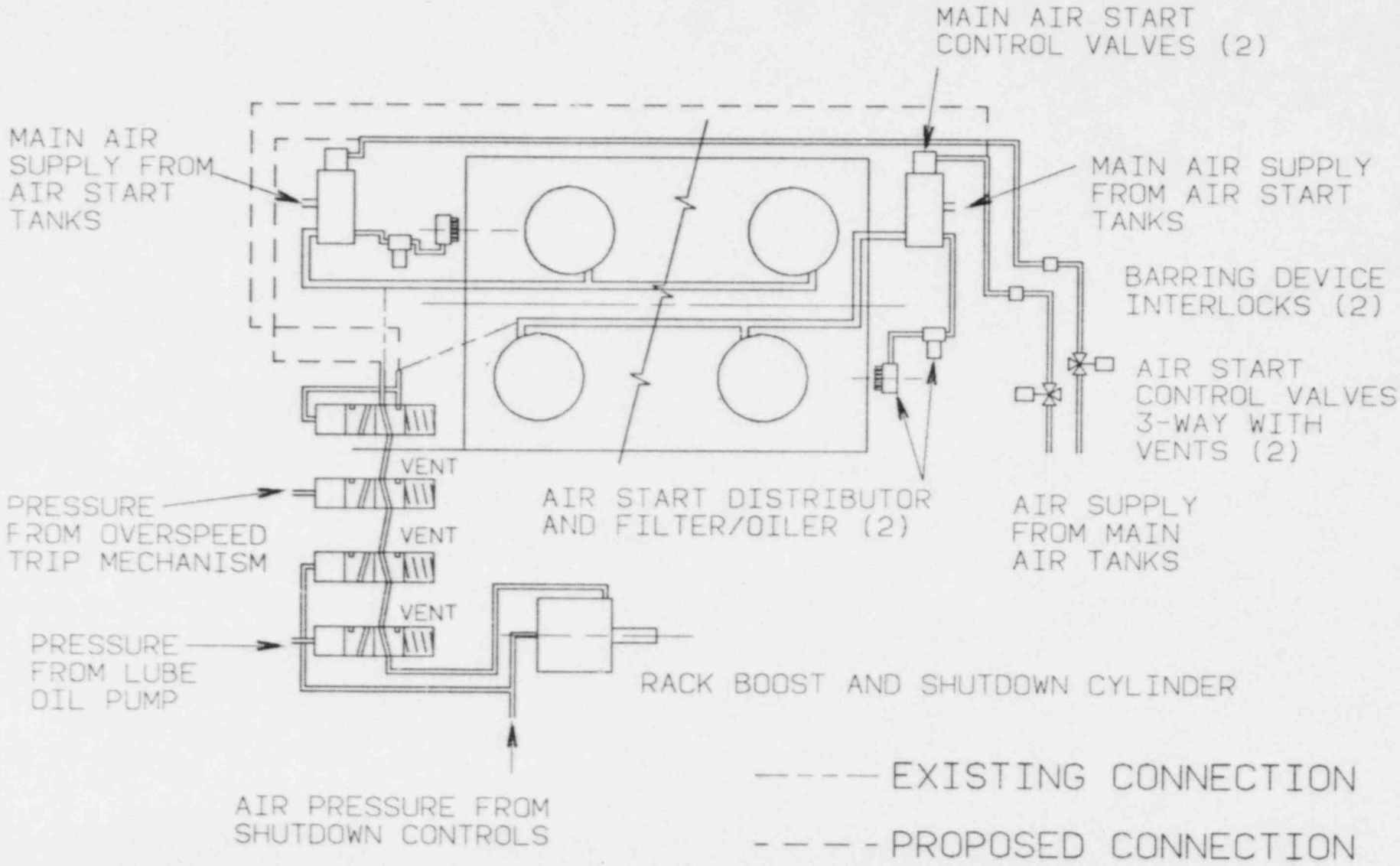


DIAGRAM OF RACK BOOST CYLINDER CONTROL CONNECTIONS

COMBUSTION ENGINEERING

STN 50-470F

February 4, 1986
LD-86-007

Mr. Harold H. Denton
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: CESSAR-F Amendment 11

Dear Mr. Denton:

Combustion Engineering, Inc. (C-E) hereby submits seventy (70) copies of Amendment No. 11 to CESSAR-F. The revisions which constitute Amendment No. 11 are summarized in the attachment for your information. These changes have resulted from C-E's response to the Severe Accident Policy Statement, NRC questions, and the CESSAR Standard Technical Specifications certification effort.

Should you have any questions or comments, please feel to call me or Mrs. R. O. Hoogewerff of my staff at (203) 285-5217.

Very truly yours,

COMBUSTION ENGINEERING, INC.

G A Davis for

A. E. Scherer
Director
Nuclear Licensing

AES/bkm
Attachment
cc: F. J. Miraglia (NRC)
G. W. Knighton (NRC)
J. H. Wilson (NRC)

B001
1/70



**Power Systems**
Combustion Engineering, Inc.

1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095-0500

(203) 688-1911
Telex: 99297

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

| | | |
|---|---|---|
| In the Matter of: | ) | |
| | ) | |
| Combustion Engineering, Inc. | ) | DOCKET NO. STN 50-470F |
| | ) | |
| Standard Plant | ) | |

APPLICATION FOR REVIEW OF
"COMBUSTION ENGINEERING STANDARD
SAFETY ANALYSIS REPORT,"
AMENDMENT NO. 11

S. T. Brewer, being duly sworn, states that he is Senior Vice President, Nuclear Power Systems, Combustion Engineering, Inc.; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this Amendment; and that all statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

COMBUSTION ENGINEERING, INC.

By S T Brewer
S. T. Brewer
Senior Vice President
Nuclear Power Systems

Sworn to before me
this 31st day of January

Brigid K. Marko
Notary Public

BRIGID K. MARKO, NOTARY PUBLIC
State of Connecticut No. 73437
Commission Expires March 31, 1990

ATTACHMENT TO LD-86-007

AMENDMENT NO. 11 REVISIONS

| Description of Revision | References |
| --- | --- |
| Chapter 1, page 1.1-6, Section 1.1.3.5 entitled Severe Reactor Accidents was added to provide the appropriate interface requirements in accordance with NRC's severe accident policy statement. | The revision was submitted in C-E letter LD-85-042, A. E. Scherer to H. R. Denton, "Forward Referenceability", August 30, 1985. |
| Appendix 15D, Table 15D-1 was revised to correct a clerical error. | The revision was discussed with and verbally approved by Mr. C. Liang of NRR. |
| Chapter 16, NSSS Technical Specifications for CESSAR System 80™ incorporates the proof and review technical specifications as transmitted by the Staff and subsequent revisions as documented in the associated references. | Revisions to the technical specifications are documented in NRC letter C. O. Thomas to A. E. Scherer, dated August 14, 1985 and certified in C-E letter LD-85-063, S. T. Brewer to H. R. Denton, "Certification of CESSAR Standard Technical Specifications", December 20, 1985. |
| Appendix B revisions concerning the CESSAR SPDS were made to facilitate NRC close out of this issue. | These revisions were discussed in C-E letter LD-85-039, A. E. Scherer to H. L. Thompson, "CESSAR Safety Parameter Display System", August 15, 1985. |

## LISTING OF AMENDMENTS

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| 1 | February 20, 1981 |
| 2 | March 12, 1981 |
| 3 | May 28, 1981 |
| 4 | July 16, 1981 |
| 5 | October 26, 1981 |
| 6 | November 20, 1981 |
| 7 | March 31, 1982 |
| 8 | May 10, 1983 |
| 9 | February 27, 1984 |
| 10 | June 28, 1985 |
| 11 | August 30, 1985 |

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August 30, 1985

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| 1.1-6 | 1.1-6<br>1.1-7 |

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August 30, 1985

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| Table 15D-1 (Sheet 1) | Table 15D-1 (Sheet 1) |

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| Remove Chapter 16 from the "Chapter 16 cover sheet" to the last page (page 6-1) | Insert Chapter 16, from the "Chapter 16 cover sheet to the last page (6-2) |

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August 30, 1985

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F) Independence - This category contains requirements to ensure safety systems and equipment are independent to the extent necessary to prevent failure of one portion or train of a safety system from causing failure of another portion or train of the same safety system.

G) Thermal Limitations - This category contains requirements that insure that thermal limitations of safety systems and/or components will not be violated.

H) Monitoring - This category contains requirements for surveillance of the safety related aspects of system performance.

I) Operational/Controls - This category contains requirements for actuation of safety related systems and components and their subsequent operations.

J) Inspection and Testing - This category contains requirements concerning demonstration of safety system operation, inservice testing, and inspection.

K) Chemistry/Sampling - This category contains requirements for safety system fluid purity and sampling.

L) Materials - This category contains material requirements for the safety systems, which include material types, special process controls, and requirements for materials in contact with primary system fluids.

M) System/Component Arrangement - This category contains special location requirements which safety systems and components place on plant arrangement in order for them to function properly.

N) Radiological Waste - This category contains requirements for radiological waste collection as a result of CESSAR design scope system operations.

O) Overpressure Protection - This category contains requirements for ensuring that CESSAR design scope safety system pressure limits are not exceeded.

P) Related Service - This category contains supporting requirements imposed on auxiliary systems by safety systems to assure they perform their safety functions.

Q) Environmental - This category contains requirements for environmental conditions which must be assured to support proper operation of CESSAR design scope safety systems and components.

R) Mechanical Interaction Between Components - This category contains requirements for consideration of differential motion, including thermal expansion and seismic effect between CESSAR design scope components and BOP components and structures.

1.1.3.3.2 Point Interfaces

The physical connections between CESSAR design scope systems and the BOP are designated point interfaces. They are shown by the "flags" on the CESSAR design scope system piping and instrumentation diagrams (P&ID's). Point interfaces are defined in Categories A, G, H, I, K, N, O, and P of the individual system interface requirements sections.

1.1.3.4 Identification of Relevant Guidance Documents

Each system's interface section includes a listing of General Design Criteria (GDC), Regulatory Guides, industry codes and standards which C-E considers relevant to the design of the base system. The GDCs, Regulatory Guides, codes and standards are not imposed as interface requirements unless specifically identified as such in the individual CESSAR/BOP interface categories.

1.1.3.5 Severe Reactor Accidents

For those System 80 plants that did not reference the CESSAR preliminary design approval in application for a construction permit, the criteria and procedural requirements stated in the 1985 NRC policy statement on severe reactor accidents must be met. These requirements are:

a. Demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements for new plants as reflected in the CP Rule [10 CFR 50.34(f)];

b. Demonstration of technical resolution of all applicable Unresolved Safety Issues and the medium-and high-priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems; 11

c. Completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that may add to the assurance of no undue risk to public health and safety; and

d. Completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgement complemented by PRA.

1.1.4 CESSAR ORGANIZATION

CESSAR is organized to respond to the Regulatory Guide 1.70, Revision 2, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", September, 1975.

### 1.1.5 SITE RELATED INFORMATION

Site dependent information is provided to the maximum extent practicable for the design of the 3817 Mwt C-E NSSS. This site dependent information, where it affects NSSS design and safety analyses, is assumed as a series of envelopes for certain site-related parameters (see Chapter 2.0) which will encompass most of the current nuclear power plant locations in the the continental United States. Non-nuclear steam supply system information or details related to a specific and/or number of sites is deferred to the applicant's safety analysis report, unless otherwise noted in CESSAR.

### 1.1.6 PIPING AND INSTRUMENTATION SYMBOLS

Figure 1.1-1 and Table 1.1-1 provides a summary of the standard symbols used for the engineering flow, piping and instrumentation diagrams, and valve lists presented in CESSAR. Table 1.1-2 provides a cross reference list between CESSAR figure numbers and C-E piping and instrumentation diagrams (P&ID) numbers.

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SEQUENCE OF EVENTS FOR A STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER AND STUCK OPEN ADV

| Time (Sec) | Event | Setpoint or Value | Success Path | |
|---|---|---|---|---|
| 0.0 | Tube Rupture Occurs | --- | | 9 |
| 40 | Third Charging Pump Started, feet below program level | -0.75 | Primary System Integrity | |
| 40 | Letdown Control Valve Throttled Back to Minimum Flow, feet below program level | -0.75 | Primary System Integrity | |
| 47 | CPC Hot Leg Saturation Trip Signal | --- | Reactivity Control | |
| 47.15 | Trip Breakers Open | --- | Reactivity Control | |
| 48 | Turbine/Generator Trip | ---<br>--- | Secondary System Integrity<br>Reactivity Control | 10 |
| 51 | Loss of Offsite Power | --- | | |
| 52 | LH Main Steam Safety Valves open, psia | 1265 | Secondary System Integrity | 9 |
| 52 | RH Main Steam Safety Valves open, psia | 1265 | Secondary System Integrity | |
| 56 | Maximum Steam Generator Pressures Both Steam Generator, psia | 1330 | | |
| 97 | Main Steam Safety Valves Closed, psia | 1190 | Secondary System Integrity | 11 |
| 121.0 | Steam Generator Water Level Reaches Emergency Feedwater Actuation Signal (EFAS) Analysis Setpoint in the Unaffected Generator, percent wide range | 25 | Secondary System Integrity | |
| 122.0 | EFAS Generated | --- | | 10 |
| 131.0 | Steam Generator Water Level Reaches EFAS Analysis Setpoint in the Affected Generator, percent wide range | 25 | Secondary System Integrity | |

TABLE 15D-1 (Page 2 of 2)

| Time (Sec) | Event | Setpoint or Value | Success Path | |
|---|---|---|---|---|
| 132.0 | EFAS Generated | --- | | |
| 167.0 | Emergency Feedwater Initiated to Unaffected Steam Generator | --- | Secondary System Integrity | 10 |
| 177.0 | Emergency Feedwater Initiated to affected Steam Generator | --- | Secondary System Integrity | |
| 460 | Operator Initiates Plant Cooldown by Opening One ADV on each SG | --- | Reactor Heat Removal | 9 |
| 546 | Pressurizer Empties | --- | | |
| 570 | Pressurizer Pressure Reaches Safety Injection Actuation Signal (SIAS) Analysis Setpoint, psia | 1578 | Reactivity Control | 10 |
| 570 | Safety Injection Actuation Signal Generated | | Reactivity Control | |
| 570 | Safety Injection Flow Initiated | --- | Reactivity Control | |
| 2100 | Operator Attempts to Isolate the Damaged Generator, RCS Tem., °F | 550 | Secondary System Integrity | |
| 3900 | Operator Closes the ADV Block Valve | --- | Secondary System Integrity | 9 |
| 4020 | Operator Initiates Auxiliary Spray Flow | | Primary System Inventory | |
| 4500 | Operator Controls Auxiliary Spray Flow, Backup Pressurizer Heater Output, and HPSI Flow to Reduce RCS Pressure and Control Subcooling, °F | 20 | Primary System Integrity | |
| 28,800 | Shutdown Cooling Entry Conditions Reached, RCS Pressure, psia/ Temperature, °F | 400/350 | Reactor Heat Removal | |

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CHAPTER 16

NSSS TECHNICAL SPECIFICATIONS

CESSAR SYSTEM 80

DOCKET NO. 50-470

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Chapter 16 is completely replaced in Amendment 11. Due to extensive format revisions, previous amendments were considered to be interim documents. Revision lines, therefore, are not shown on the revised pages of Amendment 11.

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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 specification which prescribes remedial measures required under designated conditions.

### AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

### AZIMUTHAL POWER TILT - $T_q$

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

### CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

### CHANNEL CHECK

1.5 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.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.
c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY including alarm and/or trip functions.
d. Radiological effluent process monitoring channels - See Applicant's SAR.

The CHANNEL FUNCTIONAL TEST shall include adjustment, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy.

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.
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 Not Applicable.

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 (microcuries/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."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES 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.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE SYSTEM

1.14 See Applicant's SAR.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage into closed systems, other than reactor coolant pump controlled bleed-off flow, 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 system.

MEMBER(S) OF THE PUBLIC

1.16 See Applicant's SAR.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 See Applicant's SAR.

OPERABLE - OPERABILITY

1.18 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 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and cold leg reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 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.

PLANAR RADIAL PEAKING FACTOR - $F_{xy}$

1.21 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.22 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 (PCP)

1.23 See Applicant's SAR.

PURGE - PURGING

1.24 See Applicant's SAR.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of (3800) MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 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 electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.28 See Applicant's SAR.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 See Applicant's SAR.

SOFTWARE

1.31 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOLIDIFICATION

1.32 See Applicant's SAR.

SOURCE CHECK

1.33 See Applicant's SAR.

STAGGERED TEST BASIS

1.34 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

1.35 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.36 UNIDENTIFIED LEAKAGE shall be all leakage which does not constitute either IDENTIFIED LEAKAGE or reactor coolant pump controlled bleed-off flow.

UNRESTRICTED AREA

1.37 See Applicant's SAR.

VENTILATION EXHAUST TREATMENT SYSTEM

1.38 See Applicant's SAR.

VENTING

1.39 See Applicant's SAR.

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. |

TABLE 1.2

OPERATIONAL MODES

| OPERATIONAL MODE | REACTIVITY CONDITION, $K_{eff}$ | % OF RATED THERMAL POWER* | COLD LEG TEMPERATURE ($T_{cold}$) |
|---|---|---|---|
| 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° > $T_{cold}$ > 210°F |
| 5. COLD SHUTDOWN | < 0.99 | 0 | $\leq$ 210°F |
| 6. REFUELING** | $\leq$ 0.95 | 0 | $\leq$ 135°F |

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

# 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

### 2.1.1 REACTOR CORE

DNBR

2.1.1.1 The calculated DNBR of the reactor core shall be maintained greater than or equal to 1.231.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the calculated DNBR of the reactor has decreased to less than 1.231, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kW/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kW/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, 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 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation 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:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, 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 trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| I. TRIP GENERATION | | |
| A. Process | | |
| 1. Pressurizer Pressure - High | * | * |
| 2. Pressurizer Pressure - Low(2) | * | * |
| 3. Steam Generator Level - Low(4) | * | * |
| 4. Steam Generator Level - High(9) | * | * |
| 5. Steam Generator Pressure - Low(3) | * | * |
| 6. Containment Pressure - High | * | * |
| 7. Reactor Coolant Flow - Low(7) | | |
| a. Rate(6) | * | * |
| b. Floor(6) | * | * |
| c. Band(6) | * | * |
| 8. Local Power Density - High | $\leq$ 21.0 kW/ft (5) | $\leq$ 21.0 kW/ft (5) |
| 9. DNBR - Low | $\geq$ 1.231 (5) | $\geq$ 1.231 (5) |
| B. Excore Neutron Flux | | |
| 1. Variable Overpower Trip(10) | | |
| a. Rate(8) | * | * |
| b. Ceiling(8) | * | * |
| c. Band(8) | * | * |

* See Applicant's SAR. The values shall be consistent with CESSAR FSAR and the Applicant's setpoint methodology.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 2. Logarithmic Power Level - High (1) | | |
| a. Startup and Operating | * | * |
| b. Shutdown | * | * |
| C. Core Protection Calculator System | | |
| 1. CEA Calculators | Not Applicable | Not Applicable |
| 2. Core Protection Calculators | Not Applicable | Not Applicable |
| D. Supplementary Protection System | | |
| Pressurizer Pressure - High | * | * |
| II. RPS LOGIC | | |
| A. Matrix Logic | Not Applicable | Not Applicable |
| B. Initiation Logic | Not Applicable | Not Applicable |
| III. RPS ACTUATION DEVICES | | |
| A. Reactor Trip Breakers | Not Applicable | Not Applicable |
| B. Manual Trip | Not Applicable | Not Applicable |

* See Applicant's SAR

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above $10^{-4}$% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to $10^{-4}$% of RATED THERMAL POWER.

(2) In MODES 3-4, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

The approved DNBR limit is 1.231 which includes a partial rod bow penalty compensation. If the fuel burnup exceeds that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case a DNBR trip setpoint of 1.231 is allowed provided that the difference is compensated by an increase in the CPC addressable constant BERR1 as follows:

$$BERR1_{new} = BERR1_{old} \left[1 + \frac{RB - RB_o}{100} \times \frac{d\ (\%\ POL)}{d\ (\%\ DNBR)}\right]$$

where $BERR1_{old}$ is the uncompensated value of BERR1; RB is the fuel rod bow penalty in % DNBR; $RB_o$ is the fuel rod bow penalty in % DNBR already accounted for in the DNBR limit; POL is the power operating limit; and d (% POL)/d (% DNBR) is the absolute value of the most adverse derivative of POL with respect to DNBR.

(6) RATE is the maximum rate of decrease of the trip setpoint. There are no restrictions on the rate at which the setpoint can increase.
FLOOR is the minimum value of the trip setpoint.

BAND is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor.
Setpoints are % of 100% power flow conditions.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) RATE is the maximum rate of increase of the trip setpoint. There are no restrictions on the rate at which the setpoint can decrease.
CEILING is the maximum value of the trip setpoint.
BAND is the amount by which the trip setpoint is above the input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

(10) % of RATED THERMAL POWER.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

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

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## 2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding 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 (1) 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, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kW/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operation and design basis anticipated operational occurrences is limited to 1.231 based upon a statistical combination of CE-1 CHF correlation and engineering factor uncertainties and is established as a Safety Limit. The DNBR limit of 1.231 includes a rod bow compensation of 0.8% on DNBR. For fuel burnups which exceed that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case the DNBR trip setpoint of 1.231 is allowed if the required DNBR increase is compensated by an increase of the addressable constant BERR1.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted by the CPC program.

Limiting Safety System Settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and Limiting Conditions for Operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

## 2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The reactor pressure vessel, piping, and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of the design pressure. The Reactor Coolant System valves and fittings, are designed to either Section III of the ASME Code or ANSI B 31.7, Class I, which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. See Applicant's FSAR for specific Code, Standard Editions, and Addenda. The safety limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

## 2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor 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. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Safety Limits of 1.231 and 21 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CESSAR System 80 applicable system descriptions and safety analyses.

REACTOR TRIP SETPOINTS (Continued)

Manual Reactor Trip

The Manual reactor trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above $10^{-4}$% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to $10^{-4}$% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat

Pressurizer Pressure - Low (Continued)

removal by the secondary system. During normal operation, this trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. The operator may manually bypass this trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before emergency feedwater is required to prevent degraded core cooling.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

Local Power Density - High (Continued)

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the Peak Linear Heat Rate Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of (*) psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

*See Applicant's SAR

DNBR - Low (Continued)

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the calculated core DNBR is sufficiently greater than 1.231 such that the decrease in calculated core DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

| | Parameter | Limiting Value* |
|---|---|---|
| a. | RCS Cold Leg Temperature-Low | $\geq$ °F |
| b. | RCS Cold Leg Temperature-High | $\leq$ °F |
| c. | Axial Shape Index-Positive | Not more positive than |
| d. | Axial Shape Index-Negative | Not more negative than |
| e. | Pressurizer Pressure-Low | $\geq$ psia |
| f. | Pressurizer Pressure-High | $\leq$ psia |
| g. | Integrated Radial Peaking Factor-Low | $\geq$ |
| h. | Integrated Radial Peaking Factor-High | $\leq$ |
| i. | Quality Margin-Low | $\geq$ |

Steam Generator Level - High

The Steam Generator Level - High trip provides protection in the event of excess feedwater flow. The setpoint is identical to the main steam isolation setpoint.

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two-pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary

*See Applicant's SAR

Reactor Coolant Flow - Low (continued)

side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs consistent with the safety analysis.

Pressurizer Pressure - High (SPS)

The Supplementary Protection System (SPS) augments reactor protection against overpressurization by utilizing a separate and diverse trip logic from the Reactor Protection System for initiation of reactor trip. The SPS will initiate a reactor trip when pressurizer pressure exceeds a predetermined value.

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SECTIONS 3.0 and 4.0

LIMITING COMDITIONS 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/or 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:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. 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 of 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 statements. Exceptions to these requirements are stated in the individual specifications.

## SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable 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, and

b. The combined time interval for any three consecutive surveillance intervals not to 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 have 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 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 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

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:

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

| 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 |
| 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.

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

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## 3/4.1 REACTIVITY CONTROL SYSTEMS

### 3/4.1.1 BORATION CONTROL

### SHUTDOWN MARGIN - $T_{cold}$ GREATER THAN 210°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 6.0% delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 6.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 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 6.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with $K_{eff}$ greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with $K_{eff}$ less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

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

1. Reactor Coolant System boron concentration,
2. CEA 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 ± 1.0% delta k/k 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.

## SHUTDOWN MARGIN - $T_{cold}$ LESS THAN OR EQUAL TO 210°F

### LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 4.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 4.0% delta κ/k, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 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 4.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

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

   1. Reactor Coolant System boron concentration,
   2. CEA position,
   3. Reactor Coolant System average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.

## MODERATOR TEMPERATURE COEFFICIENT

### LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation shown on Figure 3.1-1.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation shown on Figure 3.1-1, be in at least HOT STANDBY within 6 hours.

### SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 EFPD burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to two-thirds of the expected current cycle end-of-cycle core average burnup.

---

*With Keff greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

FIGURE 3.1-1
ALLOWABLE MTC MODES 1 AND 2
SEE APPLICANT'S SAR

## MINIMUM TEMPERATURE FOR CRITICALITY

### LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature ($T_{cold}$) shall be greater than or equal to 552°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature ($T_{cold}$) less than 552°F, restore $T_{cold}$ 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 ($T_{cold}$) shall be determined to be greater than or equal to 552°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 $T_{cold}$ is less than 557°F.

---

#With $K_{eff}$ greater than or equal to 1.0.

*See Special Test Exception 3.10.5.

## 3/4.1.2 BORATION SYSTEMS

### FLOW PATHS - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. If only the spent fuel pool in Specification 3.1.2.5a. is OPERABLE, a flow path from the spent fuel pool via a gravity feed connection and a charging pump to the Reactor Coolant System.

b. If only the refueling water tank in Specification 3.1.2.5b. is OPERABLE, a flow path from the refueling water tank via either a charging pump, a high pressure safety injection pump, or a low pressure safety injection pump to the Reactor Coolant System.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, 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 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.

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. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-161 (Boric Acid Filter Isolation valve) or CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°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 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 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm for 1 charging pump and 68 gpm for two charging pumps to the Reactor Coolant System.

## CHARGING PUMPS - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump* or one high pressure safety injection pump or one low pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to 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 or high pressure safety injection pump or low pressure safety injection 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

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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*Whenever the reactor coolant level is below the bottom of the pressurizer in MODE 5, one and only one charging pump shall be OPERABLE, by verifying at least once per every 7 days that power is removed from the remaining charging pumps.

## 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:

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 6% delta k/k at 210°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.

### SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

## BORATED WATER SOURCES - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

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

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

### SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1. Verifying the boron concentration of the water, and
2. Verifying the contained borated water volume of the refueling water tank or the spent fuel pool.

b. At least once per 24 hours by verifying the refueling water tank temperature when it is the source of borated water and the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when it is the source of borated water and irradiated fuel is present in the pool.

FIGURE 3.1-2
MINIMUM BORATED WATER VOLUMES
SEE APPLICANT'S SAR

FIGURE 3.1-2
MINIMUM BORATED WATER VOLUMES
SEE APPLICANT'S SAR

## BORATED WATER SOURCES - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm of boron, and
3. A solution temperature between 60°F and 120°F.

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

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool 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 6% delta k/k at 210°F, restore the above required spent fuel pool to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank 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.

### SURVEILLANCE REQUIREMENTS

4.1.2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and
2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

BORON DILUTION ALARMS

LIMITING CONDITION FOR OPERATION

3.1.2.7 Both startup channel high neutron flux alarms shall be OPERABLE.

APPLICABILITY: MODES 3*, 4, 5, and 6.

ACTION:

a. With one startup channel high neutron flux alarm inoperable:

   1. Determine the RCS boron concentration when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 through 3.1-5 by either boronometer or RCS sampling.**

b. With both startup channel high neutron flux alarms inoperable:

   1. Determine the RCS boron concentration by either boronmeter and RCS sampling** or by independent collection and analysis of two RCS samples when entering Mode 3, 4, or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 through 3.1-5, as applicable, by either boronmeter and RCS sampling** or by collection and analysis of two independent RCS samples. If redundant determination of RCS boron concentration cannot be accomplished immediately, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the method for determining and confirming RCS boron concentration is restored.

   2. When in MODE 5 with the RCS level below the centerline of the hotleg or MODE 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one startup channel high neutron flux alarm is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.7 Each startup channel high neutron flux alarm shall be demonstrated OPERABLE by performance of:

---

*Within 1 hour after the neutron flux is within the startup range following a reactor shutdown.

**With one or more reactor coolant pumps (RCP) operating the sample should be obtained from the hot leg. With no RCP operating, the sample should be obtained from the discharge line of the low pressure safety injection (LPSI) pump operating in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS (Continued)

a. A CHANNEL CHECK:

1. At least once per 12 hours.

2. When initially setting setpoints at the following times:

a) One hour after a reactor trip.
b) After a controlled reactor shutdown: Within 1 hour after the neutron flux is within the startup range in MODE 3.

b. A CHANNEL FUNCTIONAL TEST every 31 days of cumulative operation during shutdown.

TABLE 3.1-1

MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $K_{eff} > 0.98$

| OPERATIONAL MODE | Number of Operating Charging Pumps 0 | 1 | 2 | 3 |
|---|---|---|---|---|
| 3 | 12 hours | 1 hour | Operation not allowed | |
| 4 | 12 hours | 1 hour | Operation not allowed | |
| 5 RCS filled | 8 hours | 1 hour | Operation not allowed | |
| 5 RCS partially drained | Operation not allowed | | | |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

TABLE 3.1-2

MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.98 \geq K_{eff} > 0.97$

| OPERATIONAL MODE | Number of Operating Charging Pumps 0 | 1 | 2 | 3 |
|---|---|---|---|---|
| 3 | 12 hours | 2.5 hours | 1 hour | 0.5 hours |
| 4 | 12 hours | 2.5 hours | 1 hour | 0.5 hours |
| 5 RCS filled | 8 hours | 2.5 hours | 1 hour | 0.5 hours |
| 5 RCS partially drained* | 8 hours | 0.5 hours | Operation not allowed | |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

* The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

TABLE 3.1-3

MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.97 \geq K_{eff} > 0.96$

| OPERATIONAL MODE | Number of Operating Charging Pumps 0 | 1 | 2 | 3 |
|---|---|---|---|---|
| 3 | 12 hours | 3.5 hours | 1.5 hours | 1 hour |
| 4 | 12 hours | 3.5 hours | 1.5 hours | 1 hour |
| 5 RCS filled | 8 hours | 3.5 hours | 1.5 hours | 1 hour |
| 5 RCS partially drained* | 8 hours | 1 hour | Operation not allowed | |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

*The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

TABLE 3.1-4

MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $0.96 \leq K_{eff} > 0.95$

| OPERATIONAL MODE | Number of Operating Charging Pumps 0 | 1 | 2 | 3 |
|---|---|---|---|---|
| 3 | 12 hours | 5 hours | 2 hours | 1 hour |
| 4 | 12 hours | 5 hours | 2 hours | 1 hour |
| 5 RCS filled | 8 hours | 5 hours | 2 hours | 1 hour |
| 5 RCS partially drained* | 8 hours | 1.5 hours | Operation not allowed | |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

* The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

TABLE 3.1-5

MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR SYSTEM 80 EXTENDED FUEL CYCLE FOR $K_{eff} \leq 0.95$

| OPERATIONAL MODE | Number of Operating Charging Pumps 0 | 1 | 2 | 3 |
|---|---|---|---|---|
| 3 | 12 hours | 6 hours | 3 hours | 1.5 hours |
| 4 | 12 hours | 6 hours | 3 hours | 1.5 hours |
| 5 RCS filled | 8 hours | 6 hours | 3 hours | 1.5 hours |
| 5 RCS partially drained* | 8 hours | 2 hours | Operation not allowed | |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

* The Technical Specification 3.1.2.3 will allow operation of only ONE charging pump during this MODE and plant condition.

## 3/4.1.3 MOVABLE CONTROL ASSEMBLIES

### CEA POSITION

### LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length (shutdown and regulating) CEAs, and all part-length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 6.6 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length CEAs 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 at least HOT STANDBY within 6 hours.

b. With more than one full-length or part-length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one or more full-length or part-length CEAs misaligned from any other CEAs in its group by more than 6.6 inches, operation in MODES 1 and 2 may continue, provided that within 1 hour the misaligned CEA(s) is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA(s) inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA(s) shall be aligned to within 6.6 inches of the inoperable CEA(s) while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-3; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

---

*See Special Test Exceptions 3.10.2 and 3.10.4.

ACTION: (Continued)

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

Otherwise, be in at least HOT STANDBY within 6 hours.

d. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits (Figure 3.1-3) but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

e. With one part-length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 6.6 inches (indicated position) of all other part-length CEAs in its group.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length and part-length CEA shall be determined to be within 6.6 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full-length CEA not fully inserted and each part-length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

## POSITION INDICATOR CHANNELS - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit.*

### SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that, for the same CEA, the position indicator channels agree within 5.2 inches of each other at least once per 12 hours.

---

*CEAs are fully withdrawn (Full Out) when withdrawn to at least 144.75 inches (193 steps).

## POSITION INDICATOR CHANNELS - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating or part-length CEA not fully inserted.

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

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

### SURVEILLANCE REQUIREMENTS

4.1.3.3 The above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

*With the reactor trip breakers in the closed position.

## CEA DROP TIME

### LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 4 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:

a. $T_{cold}$ greater than or equal to 552°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

### SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full-length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

## SHUTDOWN CEA INSERTION LIMIT

### LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 144.75 inches (193 steps).

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 144.75 inches (193 steps), except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Withdraw the CEA to at least 144.75 inches (193 steps), or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

### SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 144.75 inches (193 steps):

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

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

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence, and to the insertion limits## shown on Figure 3.1-3** with the CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per 18 Effective Full Power Months.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 2 hours either:

   1. Restore the regulating CEA groups to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using Figure 3.1-3.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-3 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

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

**CEAs are fully withdrawn in accordance with Figure 3.1-3 when withdrawn to at least 144.75 inches (193 steps).

##Following a reactor power cutback in which (1) Regulating Group 5 is dropped or (2) Regulating Groups 4 and 5 are dropped and for cases (1) and (2) should the remaining Regulating Groups (Group 1, 2, 3, and 4) be sequentially inserted, the Transient Insertion Limit of Figure 3.1-3 can be exceeded for up to 2 hours. Also for cases (1) and (2), the specified overlap between Regulating Groups 3, 4 and 5 can be exceeded for up to 2 hours.

ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per 18 Effective Full Power Months, either:

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within 2 hours, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

FIGURE 3.1-3

CEA INSERTION LIMITS VS THERMAL POWER
SEE APPLICANT'S SAR

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## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed 14.0 kW/ft.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the linear heat rate exceeding its limits, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kW/ft or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on all OPERABLE Local Power Density channels, is less than or equal to 14.0 kW/ft.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on 14.0 kW/ft.

## 3/4.2.2 PLANAR RADIAL PEAKING FACTORS - $F_{xy}$

### LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS ($F^m_{xy}$) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS ($F^c_{xy}$) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With an $F^m_{xy}$ exceeding a corresponding $F^c_{xy}$, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to $F^m_{xy}/F^c_{xy}$ and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0]$ x 100% is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS ($F^c_{xy}$) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS ($F^m_{xy}$) or

c. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER.

### SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS ($F^m_{xy}$) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS ($F^c_{xy}$), used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

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

*See Special Test Exception 3.10.2.

## 3/4.2.3 AZIMUTHAL POWER TILT - $T_q$

### LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT ($T_q$) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but less than or equal to 0.10, within 2 hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10:

   1. Due to misalignment of either a part-length or full-length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

   2. Verify that the AZIMUTHAL POWER TILT 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 verify that the Variable Overpower Trip Setpoint has been reduced as appropriate within the next 4 hours.

   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 AZIMUTHAL POWER TILT 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 Exception 3.10.2.

## SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 EFPD to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

## 3/4.2.4 DNBR MARGIN

### LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the Region of Acceptable Operation of Figure 3.2-1 or 3.2-2, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel below the DNBR limit, within 15 minutes initiate corrective action to restore either the DNBR core power operating limit or the DNBR margin to within the limits and either:

a. Restore the DNBR core power operating limit or DNBR margin to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

### SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR margin, as indicated on all OPERABLE DNBR margin channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

4.2.4.4 The following DNBR or equivalent penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 EFPD.

| Burnup $\left(\frac{GWD}{MTU}\right)$ | DNBR Penalty (%)* |
|---|---|
| 0-10 | 0.5 |
| 10-20 | 1.0 |
| 20-30 | 2.0 |
| 30-40 | 3.5 |
| 40-50 | 5.5 |

*The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

FIGURE 3.2-1

DNBR MARGIN OPERATING LIMIT BASED ON COLSS
SEE APPLICANT'S SAR

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATOR
(COLSS OUT OF SERVICE)
SEE APPLICANT'S SAR

## 3/4.2.5 RCS FLOW RATE

### LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to $164.0 \times 10^6$ lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than or equal to its limit at least once per 12 hours.

## 3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

### LIMITING CONDITION FOR OPERATION

3.2.6 The Reactor Coolant Cold Leg Temperature ($T_c$) shall be within the Area of Acceptable Operation shown in Figure 3.2-3.

APPLICABILITY: MODE 1* and 2*

ACTION:

With the reactor coolant cold leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or be in HOT STANDBY within the next 6 hours.

### SURVEILLANCE REQUIREMENTS

4.2.6 The reactor coolant cold leg temperature shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.4.

FIGURE 3.2-3
REACTOR COOLANT COLD LEG TEMPERATURE vs CORE POWER LEVEL

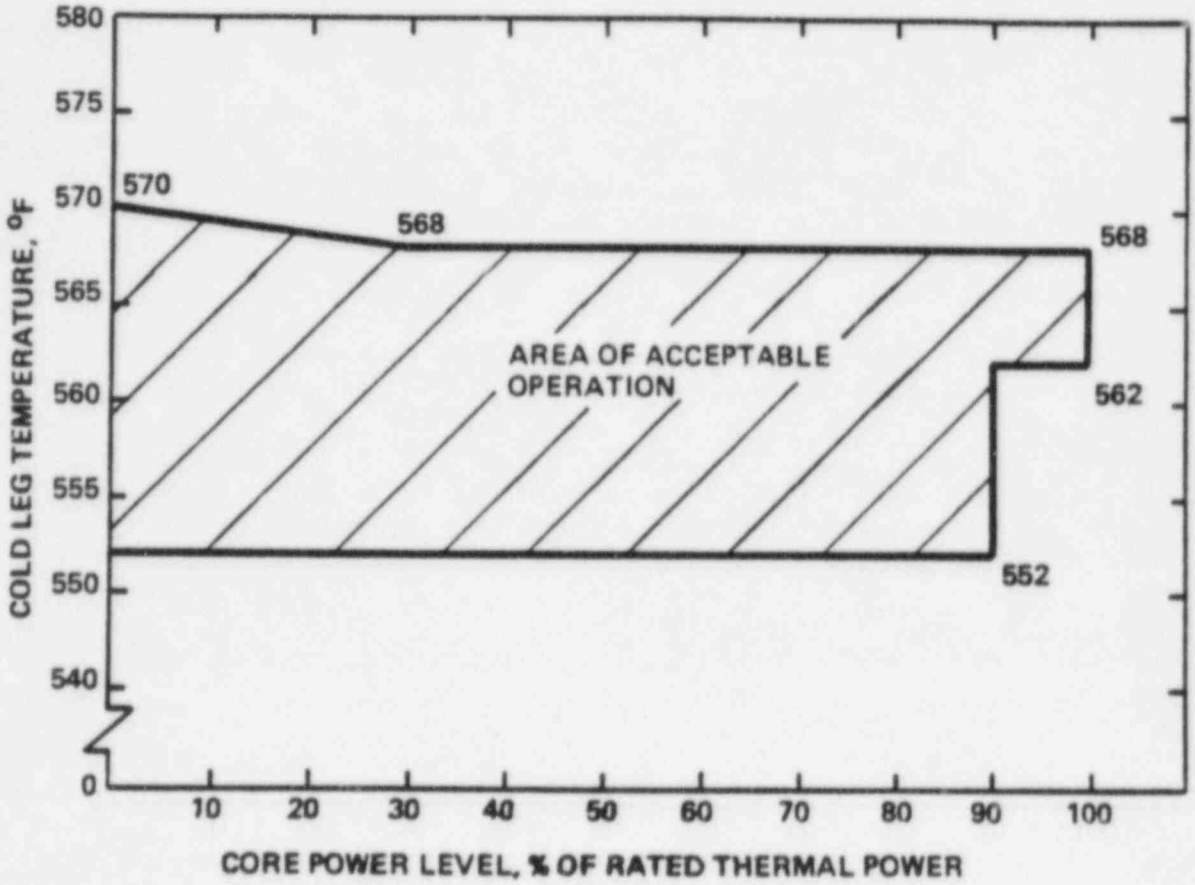


FIGURE 3.2-3

REACTOR COOLANT COLD LEG TEMPERATURE VS CORE POWER LEVEL

## 3/4.2.7 AXIAL SHAPE INDEX

### LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq ASI \leq 0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI \leq 0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*.

ACTION:

With the core average AXIAL SHAPE INDEX outside its above limits, restore the core average ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

## 3/4.2.8 PRESSURIZER PRESSURE

### LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 1815 psia and 2370 psia.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With the pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or be in at least HOT STANDBY within the next 6 hours.

### SURVEILLANCE REQUIREMENTS

4.2.8 The pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.5

## 3/4.3 INSTRUMENTATION

### 3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses 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 protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 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 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.

4.3.1.4 The isolation characteristics of each CEA isolation amplifier shall be verified at least once per 18 months during the shutdown per the following tests for the CEA position isolation amplifiers:

a. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not change by more than 0.015 volt D.C. with an applied input voltage of 5-10 volts D.C.

SURVEILLANCE REQUIREMENTS (Continued)

b. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15 volts D.C.

4.3.1.5 The Core Protection Calculators shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours. The auto restarts periodic tests Restart (Code 30) and Normal System Load (Code 33) shall not be included in this total.

4.3.1.6 The Core Protection Calculators shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1. TRIP GENERATION | | | | | |
| A. Process | | | | | |
| 1. Pressurizer Pressure - High | 4 | 2 | 3 | 1, 2 | 2#, 3# |
| 2. Pressurizer Pressure - Low | 4 | 2 (b) | 3 | 1, 2, 3*, 4* | 2#, 3# |
| 3. Steam Generator Level - Low | 4/SG | 2/SG | 3/SG | 1, 2 | 2#, 3# |
| 4. Steam Generator Level - High | 4/SG | 2/SG | 3/SG | 1, 2 | 2#, 3# |
| 5. Steam Generator Pressure - Low | 4/SG | 2/SG | 3/SG | 1, 2, 3*, 4* | 2#, 3# |
| 6. Containment Pressure - High | 4 | 2 | 3 | 1, 2 | 2#, 3# |
| 7. Reactor Coolant Flow - Low | 4/SG | 2/SG | 3/SG | 1, 2 | 2#, 3# |
| 8. Local Power Density - High | 4 | 2 (c)(d) | 3 | 1, 2 | 2#, 3# |
| 9. DNBR - Low | 4 | 2 (c)(d) | 3 | 1, 2 | 2#, 3# |
| B. Excore Neutron Flux | | | | | |
| 1. Variable Overpower Trip | 4 | 2 | 3 | 1, 2 | 2#, 3# |
| 2. Logarithmic Power Level - High | | | | | |
| a. Startup and Operating | 4 | 2 (a)(d) | 3 | 1, 2 | 2#, 3# |
| | 4 | 2 | 3 | 3*, 4*, 5* | 8 |
| b. Shutdown | 4 | 0 | 2 | 3, 4, 5 | 4 |
| C. Core Protection Calculator System | | | | | |
| 1. CEA Calculators | 2 | 1 | 2 (e) | 1, 2 | 6, 7 |
| 2. Core Protection Calculators | 4 | 2 (c)(d) | 3 | 1, 2 | 2#, 3#, 7 |

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| D. Supplementary Protection System | | | | | |
| Pressurizer Pressure - High | 4 (f) | 2 | 3 | 1, 2 | 8 |
| II. RPS LOGIC | | | | | |
| A. Matrix Logic | 6 | 1 | 3 | 1, 2 | 1 |
| | 6 | 1 | 3 | 3*, 4*, 5* | 8 |
| B. Initiation Logic | 4 | 2 | 4 | 1, 2 | 5 |
| | 4 | 2 | 4 | 3*, 4*, 5* | 8 |
| III. RPS ACTUATION DEVICES | | | | | |
| A. Reactor Trip Breaker | 4 (f) | 2 | 4 | 1, 2 | 5 |
| | 4 (f) | 2 | 4 | 3*, 4*, 5* | 8 |
| B. Manual Trip | 4 (f) | 2 | 4 | 1, 2 | 5 |
| | 4 (f) | 2 | 4 | 3*, 4*, 5* | 8 |

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above $10^{-4}$% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to $10^{-4}$% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by 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/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5 of Administrative Controls.** The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

** The Plant Review Board (PRB) shall have the responsibility for the review and documentation of judgement concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PRB meeting.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

| Process Measurement Circuit | Functional Unit Bypassed/Tripped |
|---|---|
| 1. Excore Nuclear Instrument-Linear Power (Subchannel or Linear) | Variable Overpower (RPS)<br>Local Power Density - High (RPS)<br>DNBR - Low (RPS) |
| 2. Pressurizer Pressure - High (Narrow Range) | Pressurizer Pressure - High (RPS)<br>Local Power Density - High (RPS)<br>DNBR - Low (RPS) |
| 3. Steam Generator Pressure - Low | Steam Generator Pressure - Low<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 4. Steam Generator Level - Low (Wide Range) | Steam Generator Level - Low (RPS)<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 5. Core Protection Calculator | Local Power Density - High (RPS)<br>DNBR - Low (RPS) |

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

| Process Measurement Circuit | Functional Unit Bypassed/Tripped |
|---|---|
| 1. Excore Nuclear Instrument-Linear Power (Subchannel or Linear) | Variable Overpower (RPS)<br>Local Power Density - High (RPS)<br>DNBR - Low (RPS) |
| 2. Pressurizer Pressure - High (Narrow Range) | Pressurizer Pressure - High (RPS)<br>Local Power Density - High (RPS)<br>DNBR - Low (RPS) |

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

| | | |
|---|---|---|
| 3. | Steam Generator Pressure - Low | Steam Generator Pressure - Low<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 4. | Steam Generator Level - Low (Wide Range) | Steam Generator Level - Low (RPS)<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 5. | Core Protection Calculator | Local Power Density - High (RPS)<br>DNBR - Low (RPS) |

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

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

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breaker of the inoperable channel is placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, the trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group.

b. With both CEACs inoperable operation may continue provided that:

   1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to greater than or equal to 19% of RATED THERMAL POWER and the Reactor Power Cutback System is placed out of service.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

   a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore an inoperable channel to OPERABLE status within 48 hours or open an affected reactor trip breaker within the next hour.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES[1]

| FUNCTIONAL UNIT | RESPONSE TIME |
|---|---|
| I. TRIP GENERATION | |
| A. Process | |
| 1. Pressurizer Pressure - High | ≤ seconds |
| 2. Pressurizer Pressure - Low | ≤ seconds |
| 3. Steam Generator Level - Low | ≤ seconds |
| 4. Steam Generator Level - High | ≤ seconds |
| 5. Steam Generator Pressure - Low | ≤ seconds |
| 6. Containment Pressure - High | ≤ seconds |
| 7. Reactor Coolant Flow - Low | ≤ seconds |
| 8. Local Power Density - High | |
| a. Neutron Flux Power from Excore Neutron Detectors | ≤ second* |
| b. CEA Positions | ≤ second** |
| c. CEA Positions: CEAC Penalty Factor | ≤ second** |
| 9. DNBR - Low*** | |
| a. Neutron Flux Power from Excore Neutron Detectors | ≤ second* |
| b. CEA Positions | ≤ second** |
| c. Cold Leg Temperature | ≤ second## |
| d. Hot Leg Temperature | ≤ second## |
| e. Primary Coolant Pump Shaft Speed | ≤ second# |
| f. Reactor Coolant Pressure from Pressurizer | ≤ second### |
| g. CEA Positions: CEAC Penalty Factor | ≤ second** |
| B. Excore Neutron Flux | |
| 1. Variable Overpower Trip | ≤ second* |

[1] See Applicant's SAR. The values shall be consistent with CESSAR FSAR and the Applicant's setpoint methodology.

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES[1]

| FUNCTIONAL UNIT | RESPONSE TIME |
|---|---|
| 2. Logarithmic Power Level - High | |
| a. Startup and Operating | ≤ second* |
| b. Shutdown | ≤ second* |
| C. Core Protection Calculator System | |
| 1. CEA Calculators | Not Applicable |
| 2. Core Protection Calculators | Not Applicable |
| D. Supplementary Protection System | |
| Pressurizer Pressure - High | ≤ seconds |
| II. RPS LOGIC | |
| A. Matrix Logic | Not Applicable |
| B. Initiation Logic | Not Applicable |
| III. RPS ACTUATION DEVICES | |
| A. Reactor Trip Breakers | Not Applicable |
| B. Manual Trip | Not Applicable |

*Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

**The CEA position transmitters are exempt from response time testing. The response time shall be measured from the input to the CPC, CEAC or signal isolator.

***Response times are verified using CPC Response Time Test Software, and are for hardware delays only.

#The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input.

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 6 seconds.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

[1]See Applicant's SAR.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| I. TRIP GENERATION | | | | |
| A. Process | | | | |
| 1. Pressurizer Pressure - High | S | R | M | 1, 2 |
| 2. Pressurizer Pressure - Low | S | R | M | 1, 2, 3, 4 |
| 3. Steam Generator Level - Low | S | R | M | 1, 2 |
| 4. Steam Generator Level - High | S | R | M | 1, 2 |
| 5. Steam Generator Pressure - Low | S | R | M | 1, 2, 3*, 4* |
| 6. Containment Pressure - High | S | R | M | 1, 2 |
| 7. Reactor Coolant Flow - Low | S | R | M | 1, 2 |
| 8. Local Power Density - High | S | See Core Protection Calculation System | | 1, 2 |
| 9. DNBR - Low | S | See Core Protection Calculation System | | 1, 2 |
| B. Excore Neutron Flux | | | | |
| 1. Variable Overpower Trip | S | D (2, 4), M (3, 4) Q (4) | M | 1, 2 |
| 2. Logarithmic Power Level - High | S | R (4) | M and S/U (1) | 1, 2, 3, 4, 5 and * |
| C. Core Protection Calculator System | | | | |
| 1. CEA Calculators | S | R | M, R (6) | 1, 2 |
| 2. Core Protection Calculators | S | D (2, 4), R (4, 5) M (8), S (7) | M (9), R ([illegible] | 1, 2 |

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| D. Supplementary Protection System | | | | |
| Pressurizer Pressure - High | S | R | M | 1, 2 |
| II. RPS LOGIC | | | | |
| A. Matrix Logic | N.A. | N.A. | M | 1, 2, 3*, 4*, 5* |
| B. Initiation Logic | N.A. | N.A. | M | 1, 2, 3*, 4*, 5* |
| III. RPS ACTUATION DEVICES | | | | |
| A. Reactor Trip Breakers | N.A. | N.A. | M, R (10), S/U (1) | 1, 2, 3*, 4*, 5* |
| B. Manual Trip | N.A. | N.A. | M | 1, 2, 3*, 4*, 5* |

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level, the CPC delta T power and CPC nuclear power signals to agree with the calorimetric calculation if the absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

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

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differentral pressure instrumentation and the ultrasonic flow meter adjusted pump curves (conservatively compensated for measurement uncertainties) or calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

## 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 bypasses 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 channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

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

### SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.3 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 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 ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| I. SAFETY INJECTION (SIAS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Containment Pressure - High | 4 | 2 | 3 | 1, 2, 3, 4 | 13*, 14* |
| 2. Pressurizer Pressure - Low | 4 | 2 | 3 | 1, 2, 3(a), 4(a) | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3, 4 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual SIAS (Trip Buttons) | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| II. CONTAINMENT ISOLATION (CIAS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Containment Pressure - High | 4 | 2 | 3 | 1, 2, 3 | 13*, 14*, |
| 2. Pressurizer Pressure - Low | 4 | 2 | 3 | 1, 2, 3(a) | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| II. CONTAINMENT ISOLATION (Continued) | | | | | |
| 3. Manual CIAS (Trip Buttons) | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| III. CONTAINMENT SPRAY (CSAS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| Containment Pressure -- High - High | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual CSAS (Trip Buttons) | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| IV. MAIN STEAM LINE ISOLATION (MSIS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Steam Generator Pressure - Low | 4/steam generator | 2/steam generator | 3/steam generator | 1, 2, 3(b), 4(b) | 13*, 14* |
| 2. Steam Generator Level - High | 4/steam generator | 2/steam generator | 3/steam generator | 1, 2, 3, 4 | 13*, 14* |
| 3. Containment Pressure - High | 4 | 2 | 3 | 1, 2, 3, 4 | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3, 4 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual MSIS (Trip Buttons) | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| V. RECIRCULATION (RAS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| Refueling Water Storage Tank - Low | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual RAS | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| VI. EMERGENCY FEEDWATER (SG-1)(EFAS-1) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Steam Generator #1 Level - Low | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| 2. Steam Generator Δ Pressure - SG2 > SG1 | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| VI. EMERGENCY FEEDWATER (SG-1)(EFAS-1) (Continued) | | | | | |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual EFAS | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 15 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| VII. EMERGENCY FEEDWATER (SG-2)(EFAS-2) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Steam Generator #2 Level - Low | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| 2. Steam Generator Δ Pressure - SG1 > SG2 | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual EFAS | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 15 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| VIII. LOSS OF POWER (LOV) | | | | | |
| A. (Loss of Voltage) | See Applicant's SAR | | | | |
| B. (Degraded Voltage) | See Applicant's SAR | | | | |
| IX. CONTROL ROOM EMERGENCY AIR CLEANUP | See Applicant's SAR | | | | |

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

(a) In MODES 3-4#, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-4, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

(d) The proper two-out-of-four combination.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 - 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 the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5 of Administrative Controls.** The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

| Process Measurement Circuit | |
|---|---|
| 1. Steam Generator Pressure - Low | Steam Generator Pressure - Low (RPS)<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 2. Steam Generator Level (Wide Range) | Steam Generator Level - Low (RPS)<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |

**The Plant Review Board (PRB) shall have the responsibility for the review and documentation of judgement concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PRB meeting.

# This applies only to MODE 3 for Containment Isolation (CIAS).

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

| Process Measurement Circuit | Functional Unit Bypassed/Tripped |
|---|---|
| 1. Steam Generator Pressure - Low | Steam Generator Pressure - Low (RPS)<br>Steam Generator Level 1 - Low (ESF)<br>Steam Generator Level 2 - Low (ESF) |
| 2. Steam Generator Level - Low (Wide Range) | Steam Generator Level - Low (RPS)<br>Steam Generator Level 1 - Low (ESF)<br>Steam Generator Level 2 - Low (ESF) |

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 14 are satisfied.

ACTION 15 - 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 HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, 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 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Number of Channels, 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.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES*

| ESFA SYSTEM FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| I. SAFETY INJECTION (SIAS) | | |
| A. Sensor/Trip Units | | |
| 1. Containment Pressure - High | ≤ psig | ≤ psig |
| 2. Pressurizer Pressure - Low | ≥ psia(1) | ≥ psia(1) |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| II. CONTAINMENT ISOLATION (CIAS) | | |
| A. Sensor/Trip Units | | |
| 1. Containment Pressure - High | ≤ psig | ≤ psig |
| 2. Pressurizer Pressure - Low | ≥ psia(1) | ≥ psia(1) |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| III. CONTAINMENT SPRAY (CSAS) | | |
| A. Sensor/Trip Units | | |
| Containment Pressure High - High | ≤ psig | ≤ psig |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| IV. MAIN STEAM LINE ISOLATION (MSIS) | | |
| A. Sensor/Trip Units | | |
| 1. Steam Generator Pressure - Low | ≥ psia(3) | ≥ psia(3) |
| 2. Steam Generator Level - High | ≤ NR(2) | ≤ NR(2) |
| 3. Containment Pressure - High | ≤ psig | ≤ psig |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES*

| ESFA SYSTEM FUNCTIONAL UNIT | TRIP VALUES | ALLOWABLE VALUES |
|---|---|---|
| V. RECIRCULATION (RAS) | | |
| A. Sensor/Trip Units | | |
| Refueling Water Storage Tank - Low | ≥ % of Span | ≥ % of Span ≥ |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation System | Not Applicable | Not Applicable |
| VI. EMERGENCY FEEDWATER (SG-1)(EFAS-1) | | |
| A. Sensor/Trip Units | | |
| 1. Steam Generator #1 Level - Low | ≥ % WR(4) | ≥ % WR(4) |
| 2. Steam Generator Δ Pressure - SG2 > SG1 | ≤ psid | ≤ psid |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| VII. EMERGENCY FEEDWATER (SG-2)(EFAS-2) | | |
| A. Sensor/Trip Units | | |
| 1. Steam Generator #2 Level - Low | ≥ % WR(4) | ≥ % WR(4) |
| 2. Steam Generator Δ Pressure - SG1 > SG2 | ≤ psid | ≤ psid |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| VIII. LOSS OF POWER** | | |
| A. (Loss of Voltage) | ≥ volts | ≥ volts |
| B. (Degraded Voltage) | to volts with a -second maximum time delay | to volts with a -second maximum time delay |
| IX. CONTROL ROOM EMERGENCY AIR CLEANUP** | ≤ μCi/cc | ≤ μCi/cc |

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

(1) In MODES 3-4#, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(2) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

*See Applicant's SAR. The values shall be consistent with CESSAR FSAR and the Applicant's setpoint methodology.

**Balance of Plant (BOP)

# This applies only to MODE 3 for Containment Isolation (CIAS).

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS | |
|---|---|---|
| 1. Manual | | |
| a. SIAS | | |
| Safety Injection (ECCS) | Not Applicable | |
| Containment Isolation | Not Applicable | (BOP) |
| Containment Purge Valve Isolation | Not Applicable | (BOP) |
| b. CSAS | | |
| Containment Spray | Not Applicable | (BOP) |
| c. CIAS | | |
| Containment Isolation | Not Applicable | (BOP) |
| d. MSIS | | |
| Main Steam Isolation | Not Applicable | (BOP) |
| e. RAS | | |
| Containment Sump Recirculation | Not Applicable | (BOP) |
| f. EFAS | | |
| Emergency Feedwater Pumps | Not Applicable | (BOP) |
| g. CCAS | | |
| Containment Cooling | Not Applicable | (BOP) |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES[1]

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 2. Pressurizer Pressure - Low | |
| a. Safety Injection (HPSI) | ≤ */ ** |
| b. Safety Injection (LPSI) | ≤ */ ** |
| c. Containment Isolation | ≤ */ ** |
| 3. Containment Pressure - High | |
| a. Safety Injection (HPSI) | ≤ */ ** |
| b. Safety Injection (LPSI) | ≤ */ ** |
| c. Containment Isolation | ≤ */ ** |
| d. Main Steam Isolation | ≤ */ ** |
| e. Containment Spray Pump | ≤ */ ** |
| 4. Containment Pressure - High-High | |
| a. Containment Spray | ≤ */ ** |
| 5. Steam Generator Pressure - Low | |
| a. Main Steam Isolation | ≤ */ ** |
| 6. Refueling Water Tank - Low | |
| a. Containment Sump Recirculation | ≤ */ ** |
| 7. Steam Generator Level - Low | |
| a. Emergency Feedwater | ≤ */ ** |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES[1]

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 8. Steam Generator Level - High | |
| a. Main Steam Isolation | ≤ */ ** |
| 9. Steam Generator ΔP-High-Coincident With Steam Generator Level Low | |
| a. Emergency Feedwater Isolation from the Ruptured Steam Generator | ≤ */ ** |
| 10. Control Room Essential Filtration Actuation[2] | ≤ */≤ ** |
| 11. (Degraded Voltage)[2] | |
| Loss of Power 90% system voltage | ≤ |
| 12. (loss of Voltage)[2] | |
| Loss of Power | ≤ |

TABLE NOTATIONS

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

[1]See Applicant's SAR. The values shall be consistent with CESSAR FSAR and the Applicant's setpoint methodology.

[2]Balance of Plant (BOP)

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| I. SAFETY INJECTION (SIAS) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Containment Pressure - High | S | R | M | 1, 2, 3, 4 |
| 2. Pressurizer Pressure - Low | S | R | M | 1, 2, 3, 4 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | NA | NA | M | 1, 2, 3, 4 |
| 2. Initiation Logic | NA | NA | M | 1, 2, 3, 4 |
| 3. Manual SIAS | NA | NA | M | 1, 2, 3, 4 |
| C. Automatic Actuation Logic | NA | NA | M(1) (2) (3) | 1, 2, 3, 4 |
| II. CONTAINMENT ISOLATION (CIAS) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Containment Pressure - High | S | R | M | 1, 2, 3 |
| 2. Pressurizer Pressure - Low | S | R | M | 1, 2, 3 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | NA | NA | M | 1, 2, 3, 4 |
| 2. Initiation Logic | NA | NA | M | 1, 2, 3, 4 |
| 3. Manual CIAS | NA | NA | M | 1, 2, 3, 4 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| II. CONTAINMENT ISOLATION (Continued) | | | | |
| C. Automatic Actuation Logic | NA | NA | M(1) (2) (3) | 1, 2, 3, 4 |
| III. CONTAINMENT SPRAY (CSAS) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Containment Pressure -- High - High | S | R | M | 1, 2, 3 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | NA | NA | M | 1, 2, 3, 4 |
| 2. Initiation Logic | NA | NA | M | 1, 2, 3, 4 |
| 3. Manual CSAS | NA | NA | M | 1, 2, 3, 4 |
| C. Automatic Actuation Logic | NA | NA | M(1) (2) (3) | 1, 2, 3, 4 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| IV. MAIN STEAM LINE ISOLATION (MSIS) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Steam Generator Pressure - Low | S | R | M | 1, 2, 3, 4 |
| 2. Steam Generator Level - High | S | R | M | 1, 2, 3, 4 |
| 3. Containment Pressure - High | S | R | M | 1, 2, 3, 4 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | NA | NA | M | 1, 2, 3, 4 |
| 2. Initiation Logic | NA | NA | M | 1, 2, 3, 4 |
| 3. Manual MSIS | NA | NA | M | 1, 2, 3, 4 |
| C. Automatic Acutation Logic | NA | NA | M(1) (2) (3) | 1, 2, 3, 4 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| V. RECIRCULATION (RAS) | | | | |
| A. Sensor/Trip Units | | | | |
| Refueling Water Storage Tank - Low | S | R | M | 1, 2, 3 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | NA | NA | M | 1, 2, 3, 4 |
| 2. Initiation Logic | NA | NA | M | 1, 2, 3, 4 |
| 3. Manual RAS | NA | NA | M | 1, 2, 3, 4 |
| C. Automatic Acutation Logic | NA | NA | M(1) (2) (3) | 1, 2, 3, 4 |
| VI. EMERGENCY FEEDWATER (SG-1)(EFAS-1) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Steam Generator #1 Level - Low | S | R | M | 1, 2, 3 |
| 2. Steam Generator Δ Pressure SG2 > SG1 | S | R | M | 1, 2, 3 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED | |
|---|---|---|---|---|---|
| VI. EMERGENCY FEEDWATER (SG-1)(EFAS-1) (Continued) | | | | | |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | NA | NA | M | 1, 2, 3, 4 | |
| 2. Initiation Logic | NA | NA | M | 1, 2, 3, 4 | |
| 3. Manual AFAS | NA | NA | M | 1, 2, 3, 4 | |
| C. Automatic Actuation Logic | NA | NA | M(1) (2) (3) | 1, 2, 3, 4 | |
| VII. EMERGENCY FEEDWATER (SG-2)(EFAS-2) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Steam Generator #2 Level - Low | S | R | M | 1, 2, 3 | |
| 2. Steam Generator Δ Pressure SG1 > SG2 | S | R | M | 1, 2, 3 | |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | NA | NA | M | 1, 2, 3, 4 | |
| 2. Initiation Logic | NA | NA | M | 1, 2, 3, 4 | |
| 3. Manual AFAS | NA | NA | M | 1, 2, 3, 4 | |
| C. Automatic Actuation Logic | NA | NA | M(1) (2) (3) | 1, 2, 3, 4 | |
| VIII. LOSS OF POWER (LOV) | | | | | |
| A. (Loss of Voltage) | | See Applicant's SAR | | | (BOP) |
| B. (Degraded Voltage) | | See Applicant's SAR | | | (BOP) |

TABLE 4.3-2 (Continued)

TABLE NOTATION

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

(2) Testing of automatic actuation logic shall include energization/deenergization of each initiation relay and verification of proper operation of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays listed below are exempt from testing during POWER OPERATION but shall be tested at least once per 18 months during REFUELING and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

ACTUATION DEVICES THAT CANNOT BE TESTED AT POWER

| TRAIN A | | TRAIN B | |
|---|---|---|---|
| ESF FUNCTION | ACTUATION DEVICE | ESF FUNCTION | ACTUATION DEVICE |
| ** | ** | ** | ** |

In the case of the following relays which are tested during power operation, one or more pieces of equipment cannot be actuated, but can be racked out, bypassed or etc., which will not preclude the relay from being tested but will not actuate the locked out equipment associated with the relay:

| | | | |
|---|---|---|---|
| ** | ** | ** | ** |

** See Applicant's SAR.

## 3/4.3.3 MONITORING INSTRUMENTATION

### RADIATION MONITORING INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3.1 Radiation Monitoring Instrumentation

Table 3.3-6, Radiation Monitoring Instrumentation

Table 4.3-3, Radiation Monitoring Instrumentation Surveillance Requirements

See Applicant's SAR.

## INCORE DETECTORS

### LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

a. With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

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

### SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if the system has just been returned to OPERABLE status or if 7 days or more have elasped since last use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes al electronic components. The fixed incore neutron detectors shall be calibrated prior to installation in the reactor core.

## SEISMIC AND METEOROLOGICAL INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3.3 Seismic Monitoring Instrumentation

Table 3.3-7, Seismic Monitoring Instrumentation

Table 4.3-5, Seismic Monitoring Instrumentation Surveillance Requirements

3/4.3.3.4 Meteorological Monitoring Instrumentation

Table 3.3-8, Meteorological Monitoring Instrumentation

Table 4.3-5, Meteorological Monitoring Instrumentation Surveillance Requirements

See Applicant's SAR

## REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown system 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 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 instrumentation control circuits required by Table 3.3-9 inoperable, restore the inoperable circuit(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.3.3.5 The remote shutdown system shall be demonstrated OPERABLE:

a. By performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6 for each remote shutdown monitoring instrumentation channel.

b. By operation of each remote shutdown system instrumentation control circuit including the actuated components at least once per 18 months.

TABLE 3.3-9

REMOTE SHUTDOWN INSTRUMENTATION

| INSTRUMENT* | READOUT LOCATION | MINIMUM CHANNELS OPERABLE |
|---|---|---|
| 1. Logarithmic Neutron Channel | Remote Shutdown Panel | 1 |
| 2. Reactor Coolant Hot Leg Temperature | Remote Shutdown Panel | 1/loop |
| 3. Reactor Coolant Cold Leg Temperature | Remote Shutdown Panel | 1/loop |
| 4. Pressurizer Pressure | Remote Shutdown Panel | 1 |
| 5. Pressurizer Level | Remote Shutdown Panel | 1 |
| 6. Steam Generator Pressure | Remote Shutdown Panel | 1/steam generator |
| 7. Steam Generator Level | Remote Shutdown Panel | 1/steam generator |
| 8. Refueling Water Tank Level | Remote Shutdown Panel | 1 |
| 9. Charging Line Pressure | Remote Shutdown Panel | 1 |
| 10. Charging Line Flow | Remote Shutdown Panel | 1 |
| 11. Shutdown Cooling Heat Exchanger Temperatures | Remote Shutdown Panel | 1 |
| 12. Shutdown Cooling Flow | Remote Shutdown Panel | 1 |
| 13. Emergency Feedwater Flow Rate | Remote Shutdown Panel | 1/steam generator |

| CONTROL CIRCUIT* | SWITCH LOCATION* |
|---|---|
| 1. Emergency Generator (BOP) | |
| 2. Emergency Generator Fuel Storage and Transfer System (BOP) | |
| 3. Emergency Power Distribution System (BOP) | |
| 4. Nuclear Service Water System (BOP) | |
| 5. Component Cooling Water System (BOP) | |
| 6. Emergency Feedwater System (BOP) | |
| 7. Reactor coolant pump trip pushbuttons | |
| 8. Backup heater groups 1 and 2 | |
| 9. Atmospheric steam dump valve control switches | |
| 10. Pressurizer auxiliary spray valves controls | |
| 11. Letdown isolation valves controls | |
| 12. Reactor coolant pump seal bleed off valve controls | |

* See Applicant's SAR.

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

| CONTROL CIRCUIT* | SWITCH LOCATION* |
|---|---|
| 13. MSIS actuation pushbuttons | |
| 14. Low pressurizer pressure setpoint reset and bypass | |
| 15. LPSI pumps | |
| 16. SIT vent valves | |
| 17. SIT isolation valves | |
| 18. LPSI/CS pumps cross-connect values | |
| 19. Shutdown cooling heat exchanger intake and exit valves | |
| 20. LPSI pump mini-flow valves | |
| 21. LPSI pump suction valves | |
| 22. LPSI isolation valves | |
| 23. Shutdown cooling heat exchanger spray bypass valves | |
| 24. Shutdown cooling heat exchanger flow control valves | |
| 25. Shutdown cooling warm-up bypass valves | |
| 26. Shutdown cooling suction line valves; and | |
| 27. Shutdown heat exchanger bypass flow control valves. | |

*See applicant's SAR

TABLE 4.3-6

REMOTE SHUTDOWN INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION | |
|---|---|---|---|
| 1. Logarithmic Neutron Channel | M | R | |
| 2. Reactor Coolant Hot Leg Temperature (2) | M | R | |
| 3. Reactor Coolant Cold Leg Temperature (2) | M | R | |
| 4. Pressurizer Pressure | M | R | |
| 5. Pressurizer Level | M | R | |
| 6. Steam Generator Pressure | M | R | |
| 7. Steam Generator Level | M | R | |
| 8. Refueling Water Tank Level | M | R | |
| 9. Charging Line Pressure | M | R | |
| 10. Charging Line Flow | M | R | |
| 11. Shutdown Cooling Heat Exchanger Temperatures | M | R | |
| 12. Shutdown Cooling Flow | M | R | |
| 13. Emergency Feedwater Flow Rate | M | R | (BOP) |

## POST-ACCIDENT MONITORING INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more accident monitoring instrumentation channels inoperable, take the action shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

| INSTRUMENT | REQUIRED NUMBER OF CHANNELS | MINIMUM CHANNELS OPERABLE | ACTION |
|---|---|---|---|

See Applicant's SAR.

TABLE 3.3-10
ACTION STATEMENTS

See Applicant's SAR.

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|

See Applicant's SAR.

## CHLORINE DETECTION, FIRE DETECTION, LOOSE-PART DETECTION, AND RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.3.7 Chlorine Detection Systems

3/4.3.3.8 Fire Detection Instrumentation

Table 3.3-11, Fire Detection Instruments

3/4.3.3.9 Loose-Part Detection Instrumentation

Table 3.3-12, Loose Parts Sensor Locations

3/4.3.3.10 Radioactive Gaseous Effluent Monitoring Instrumentation

Table 3.3-13, Radioactive Gaseous Effluent Monitoring Instrumentation

Table 4.3-8, Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

See Applicant's SAR

## 3/4.3.4 TURBINE OVERSPEED PROTECTION

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.4 Turbine Overspeed Protection

See Applicant's SAR.

## 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 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exception 3.10.3.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 The reactor coolant loops listed below shall be OPERABLE and at least one of these reactor coolant loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump.

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated 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 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 to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one reactor coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE verifying the secondary side water level to be $\geq$ 25% indicated wide range level 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.

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump,**

c. Shutdown Cooling Train #1,

d. Shutdown Cooling Train #2.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling 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 coolant loop to operation.

---

*All reactor coolant pumps and shutdown cooling 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 ***°F during cooldown, or ***°F during heatup, unless the secondary water temperature (saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

***See Applicant's SAR

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined to be 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 the secondary side water level to be $\geq$ 25% indicated wide range level at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate greater than or equal to 4000 gpm at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 25% indicated wide range level.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no shutdown cooling 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 shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of both 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 shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

---

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling 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 **°F during cooldown, or **°F during heatup, unless the secondary water temperature (saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

*The shutdown cooling 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.

**See Applicant's SAR

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

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.

b. With no shutdown cooling 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 shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

---

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

*The shutdown cooling pump may be de-energized 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.

## 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 2500 psia ± 1%.*

APPLICABILITY: MODE 4

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

b. The provisions of Specification 3.0.4 may be suspended for up to 12 hours for entering into and during operation in MODE 4 for purposes of setting the pressurizer code safety valves under ambient (HOT) conditions provided a preliminary cold setting was made prior to heatup.

### SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance 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.

## LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 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 HOT SHUTDOWN within the following 6 hours with the shutdown cooling system suction line relief valves aligned to provide overpressure protection for the Reactor Coolant System.

## SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance 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.

## 3/4.4.3 PRESSURIZER

### PRESSURIZER

### LIMITING CONDITION FOR OPERATION

3.4.3.1 The pressurizer shall be OPERABLE with:

a. A steady state water volume less than or equal to 58% indicated level (996 cu. ft.) but greater than 27% indicated level (440 cu. ft.), and

b. At least two groups of pressurizer heaters capable of being powered from 1E buses each having a nominal capacity of at least 150 kw.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of the above required 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, restore the pressurizer to OPERABLE status within 1 hour, or be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.4.3.1.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.3.1.2 The capacity of the above required groups of pressurizer heaters shall be verified to be at least 150 kW at least once per 92 days.

4.4.3.1.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power:

a. The pressurizer heaters are automatically shed from the emergency power sources, and

b. The pressurizer heaters can be reconnected to their respective buses manually from the control room.

AUXILIARY SPRAY*

LIMITING CONDITION FOR OPERATION

3.4.3.2 Both auxiliary spray valves shall be OPERABLE.

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

ACTION:

a. With only one of the above required auxiliary spray valves OPERABLE, restore both valves 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 none of the above required auxiliary spray valves OPERABLE, restore at least one valve to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The auxiliary spray valves shall be verified to have power available to each valve every 24 hours.

4.4.3.2.2 The auxiliary spray valves shall be cycled at least once per 18 months.

*This technical specification may be subject to change following NRC resolution of the auxiliary spray issue for CESSAR.

## 3/4.4.4 STEAM GENERATORS

### LIMITING CONDITION FOR OPERATION

3.4.4 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 $T_{cold}$ above 210°F.

### SURVEILLANCE REQUIREMENTS

4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

4.4.4.1 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.

4.4.4.2 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. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.4.3 and the inspected tubes shall be verified acceptable per Specification 4.4.4.4. 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:

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.4.4a.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 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.

2. The inspections include those portions of the tubes where imperfections were previously found.

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.3 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 calender 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 following service under AVT conditions, 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 of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into 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 4.4.4.3a.; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled 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. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

4.4.4.4 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 manufacturing related or 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 or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service or repaired and is equal to ** of the nominal tube wall thickness.

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 4.4.4.3c., 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.

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

**See Applicant's SAR. Value to be determined in accordance with recommendations of Regulatory Guide 1.121, August 1976.

condition of the tubing. This inspection was performed prior to the field hydrostatic test and 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.

4.4.4.5 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 of Administrative Controls.

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 of Administrative Controls within 12 months following 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.

3. Identification of tubes plugged and/or repaired.

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 of Administrative Controls within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

## TABLE 4.4-1

## MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes 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.

TABLE 4.4-2

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<br>C-2 | None<br>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 |
| | C-3 | Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G.<br><br>Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50 | 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. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50 | N.A. | N.A. |

$S = 3 \frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

## 3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

### LEAKAGE DETECTION SYSTEMS

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.4.5.1 Reactor Coolant System Leakage Detection Systems

See Applicant's SAR

## OPERATIONAL LEAKAGE

### LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators, and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 ± 20 psia 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 limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within its 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 one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

### SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 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,

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.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

| VALVE* | DESCRIPTION |
|---|---|
| 1) | |
| 2) | |
| 3) | |
| 4) | |
| 5) | |
| 6) | |
| 7) | |
| 8) | |
| 9) | |
| 10) | |

* See Applicant's SAR

## 3/4.4.6 CHEMISTRY

### LIMITING CONDITION FOR OPERATION

3.4.6 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.

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 psia, 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 psia or prior to proceeding to MODE 4.

### SURVEILLANCE REQUIREMENTS

4.4.6 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.

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY

| PARAMETER | STEADY STATE LIMIT | TRANSIENT LIMIT |
|---|---|---|
| DISSOLVED OXYGEN* | ≤ 0.10 ppm | ≤ 1.00 ppm |
| CHLORIDE | ≤ 0.15 ppm | ≤ 1.50 ppm |
| FLUORIDE | ≤ 0.10 ppm | ≤ 1.00 ppm |

*Limit not applicable with $T_{cold}$ less than or equal to 250°F.

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 $T_{cold}$ less than or equal to 250°F

## 3/4.4.7 SPECIFIC ACTIVITY

### LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

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

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9 of Administrative Controls within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with $T_{cold}$ less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with $T_{cold}$ less than 500°F within 6 hours.

---

*With $T_{cold}$ greater than or equal to 500°F.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

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

d. With the specific activity of the primary coolant greater than 1 microcurie/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4.(a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9 of Administrative Controls. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of degassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY 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 Activity 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 $\bar{E}$ 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 1.0 μCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, 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. One sample is sufficient if plant has gone through a SHUTDOWN or if transient is complete in 6 hours. | 1, 2, 3 |

# Until the specific activity of the primary coolant system is restored within its limits.

* 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.

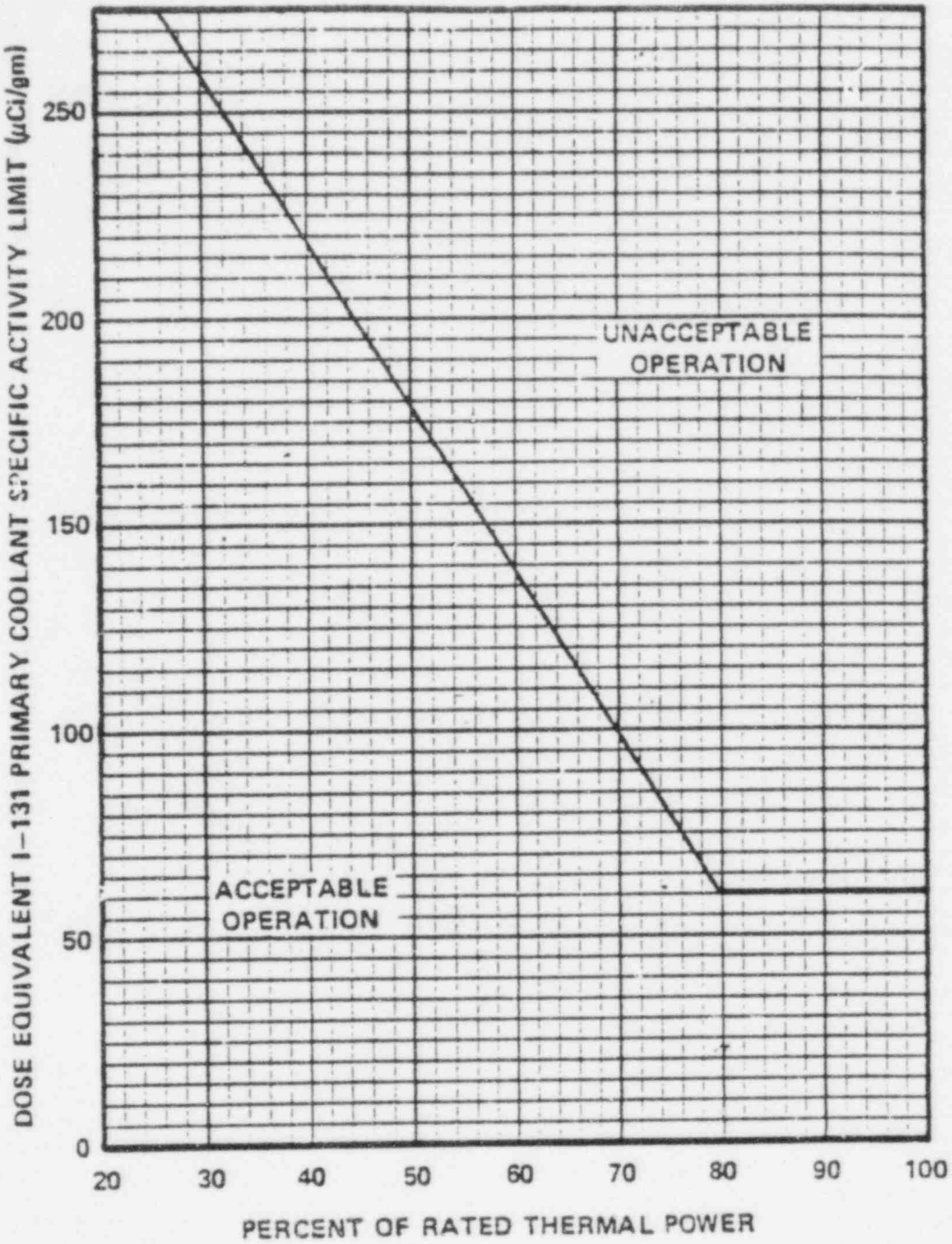


FIGURE 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131

## 3/4.4.8 PRESSURE/TEMPERATURE LIMITS

### REACTOR COOLANT SYSTEM

### LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 100°F per hour.

b. A maximum cooldown rate of 100°F per hour.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves in Figure 3.4-2.

APPLICABILITY: At all times.*

ACTION:

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 operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS $T_{cold}$ and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.4.8.1.1 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.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals 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 Figure 3.4-2.

*See Special Test Exception 3.10.5.

FIGURE 3/4 3.4-2

RCS PRESS/TEMP LIMITS (0 - 10 YRS) FULL POWER OPERATION
(SEE APPLICANT'S SAR)

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE*

| CAPSULE NUMER | VESSEL LOCATION | LEAD FACTOR | WITHDRAWAL TIME (EFPY) |
|---|---|---|---|
| 1 | 38° | 1.5 | Standby |
| 2 | 43° | 1.5 | Standby |
| 3 | 137° | 1.5 | 4 - 5 |
| 4 | 142° | 1.5 | Standby |
| 5 | 230° | 1.5 | 12 - 15 |
| 6 | 310° | 1.5 | 18 - 24 |

* Typical

## PRESSURIZER HEATUP/COOLDOWN LIMITS

### LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer temperature shall be limited to:

a. A maximum heatup rate of 200°F per hour, and

b. A maximum cooldown rate of 200°F per hour.

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

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-2 for each cycle of main spray with less than four reactor coolant pumps operating and for each cycle of auxiliary spray operation.

## OVERPRESSURE PROTECTION SYSTEMS

### LIMITING CONDITION FOR OPERATION

3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to ** psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. **°F during cooldown

b. **°F during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce $T_{cold}$ to less than 200°F and, depressurize and vent the RCS through a greater than or equal to ** square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce $T_{cold}$ to less than 200°F and, depressurize and vent the RCS through a greater than or equal to ** square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9 of Administrative Controls within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves 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.

### SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during

a. Cooldown with the RCS temperature less than or equal to **°F.

b. Heatup with the RCS temperature less than or equal to **°F.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

** See Applicant's SAR

## 3/4.4.9 STRUCTURAL INTEGRITY

### LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.9.

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) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 210°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 to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.4.9 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 **.

**See Applicant's SAR.

## 3/4.4.10 REACTOR COOLANT SYSTEM VENTS

### LIMITING CONDITION FOR OPERATION

3.4.10 Both reactor coolant system vent paths from the reactor vessel head shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With only one of the above required reactor vessel head vent paths OPERABLE, restore both paths 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 none of the above required reactor vessel head vent paths OPERABLE, restore at least one path to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.4.10 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.

b. Cycling each vent through at least one complete cycle from the control room.

c. Verifying flow through the reactor coolant system vent paths during venting.

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## 3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve key-locked open and power to the valve removed,

b. A contained borated water level of between 28% (1802 cubic feet) and 72% (1914 cubic feet)

c. A boron concentration between 4000 and 4400 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

e. Nitrogen vent valves closed and power removed.**

f. Nitrogen vent valves are capable of being operated upon restoration of power.

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

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours or in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

---

†With pressurizer pressure greater than or equal to 1750 psia. When pressurizer pressure is less than 1750 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 0% narrow range (corresponding to 60% wide range indication or 1415 cubic feet) and 72% narrow range indication (corresponding to 81% wide range indication or 1914 cubic feet). With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 0% narrow range (corresponding to 39% wide range indication or 962 cubic feet) and 72% narrow range indication (corresponding to 81% wide range indication or 1914 cubic feet). In MODE 4 with pressurizer pressure less than 430 psia, the safety injection tanks may be isolated.

*See Special Test Exceptions 3.10.7.

**Nitrogen vent valves may be cycled as necessary to maintain the required nitrogen cover pressure per Specification 3.5.1d.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each safety injection tank isolation valve is open and the nitrogen vent valves are closed.

b. At least once per 31 days and within 6 hours after each solution level increase of greater than or equal to 7% of tank narrow range level by verifying the boron concentration of the safety injection tank solution is between 4000 and 4400 ppm.

c. At least once per 31 days when the RCS pressure is above 715 psia, by verifying that power to the isolation valve operator is removed.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. When an actual or simulated RCS pressure signal exceeds 515 psia, and

2. Upon receipt of a safety injection actuation (SIAS) test signal.

e. At least once per 18 months by verifying OPERABILITY of the RCS-SIT differential pressure alarm by simulating RCS pressure > 715 psia with SIT pressure < 600 psig.

f. At least once per 18 months, when SITs are isolated, by verifying the SIT nitrogen vent valves can be opened.

g. At least once per 31 days, by verifying that power is removed from the nitrogen vent valves.

## 3/4.5.2 ECCS SUBSYSTEMS - $T_{cold}$ 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 high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

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 of Administrative Controls 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 injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 1750 psia.

SURVEILLANCE REQUIREMENTS

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 the power to the valve operators removed:

| Valve Number** | Valve Function | Valve Position |
|---|---|---|
| 1. SI-604 | 1. HOT LEG INJECTION | 1. SHUT |
| 2. SI-609 | 2. HOT LEG INJECTION | 2. SHUT |

b. At least once per 31 days by:

1. 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, and

2. Verifying that the ECCS piping is full of water by venting the ECCS pump casing and accessible discharge piping high points.

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. For all the affected areas within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

**Requirement applicable only if SI-604, SI-609, SI-321, SI-331 are not each supplied by an independent and redundant emergency power source.

SURVEILLANCE REQUIREMENTS (Continued)

1. 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 corrosion.

2. Verifying that a minimum total of **. (BOP)

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 a SIAS and RAS test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

a. High Pressure Safety Injection pump.

b. Low Pressure Safety Injection pump.

3. Verifying that on a recirculation actuation test signal, the containment sump isolation valves open, the HPSI, LPSI and CS pump minimum bypass recirculation flow line isolation valves and combined SI mini-flow valve close, and the LPSI pumps stop.

f. By verifying that each of the following pumps develops the indicated differential pressure at or greater than their respective minimum allowable recirculation flow when tested pursuant to Specification 4.0.5:

1. High pressure safety injection pump greater than or equal to (see Applicant's SAR).

2. Low pressure safety injection pump greater than or equal to (see Applicant's SAR).

** See Applicant's SAR for means of controlling the pH in the containment sump after an LOCA.

SURVEILLANCE REQUIREMENTS (Continued)

g. By verifying the correct position of each 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.

2. At least once per 18 months.

| HPSI System Valve Number | | | LPSI System Valve Number | | | Hot Leg Injection Valve Number | |
|---|---|---|---|---|---|---|---|
| 1. | SI-617, | SI-616 | 1. | SI-615, | SI-306 | 1. | SI-321 |
| 2. | SI-627, | SI-626 | 2. | SI-625, | SI-307 | 2. | SI-331 |
| 3. | SI-637, | SI-636 | 3. | SI-635 | | | |
| 4. | SI-647, | SI-646 | 4. | SI-645 | | | |

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 the following flow rates:

HPSI System - Single Pump

The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to ** gpm.

LPSI System - Single Pump

1. Injection Loop 1, total flow equal to ** gpm
2. Injection Legs 1A and 1B when tested individually, with the other leg isolated, shall be within ** gpm of each other.
3. Injection Loop 2, total flow equal to ** gpm
4. Injection Legs 2A and 2B when tested individually, with the other leg isolated, shall be within ** gpm of each other.

Simultaneous Hot Leg and Cold Leg Injection - Single Pump

1. Hot Leg, flow equal to ** gpm
2. Cold Leg, flow equal to ** gpm

** See Applicant's SAR

## 3/4.5.3 ECCS SUBSYSTEMS - $T_{cold}$ 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. An OPERABLE high pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 3* AND 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 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 of Administrative Controls 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.

### SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable surveillance requirements of Specification 4.5.2.

*With pressurizer pressure less than 1750 psia.

## 3/4.5.4 REFUELING WATER TANK

### LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank (RWT) shall be OPERABLE with:

a. A minimum borated water volume as specified in Figure 3.1-2 of Specification 3.1.2.5,

b. A boron concentration between 4000 and 4400 ppm of boron, and

c. A solution temperature between 60°F and 120°F.

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

ACTION:

With the refueling water tank 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

4.5.4 The RWT 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 RWT temperature when the (outside) air temperature is outside the 60°F to 120°F range.

## 3/4.6 CONTAINMENT SYSTEMS

## 3/4.6.1 PRIMARY CONTAINMENT

## CONTAINMENT INTEGRITY

## LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.6.1.1 Containment Integrity

3/4.6.1.2 Containment Leakage

3/4.6.1.3 Containment Air Locks

3/4.6.1.4 Containment Isolation Valve and Channel Weld Pressurization Systems (Optional)

3/4.6.1.5 Internal Pressure

3/4.6.1.6 Air Temperature

3/4.6.1.7 Containment Structural Integrity

Table 4.6-1, Tendon Surveillance -- First Year

Table 4.6-2, Tendon Lift-Off Force First Year

3/4.6.1.8 Containment Ventilation System

See Applicant's SAR

## 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 RWT on a containment spray actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal. Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

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.

### 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 is positioned to take suction from the RWT on a containment spray actuation (CSAS) test signal.

b. By verifying that each pump develops an indicated differential pressure of greater than or equal to 200 psid at greater than or equal to the minimum allowable bypass recirculation flowrate when tested pursuant to Specification 4.0.5.

c. At least once per 31 days by verifying that the system piping is full of water from the RWT to the header isolation valves.

d. 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 Containment Spray Actuation (CSAS) and Recirculation Actuation (RAS) Test Signal.
   2. Verifying that upon a Recirculation Actuation Test Signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump starts automatically on a containment spray actuation (CSAS) test signal.

e. 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.

## IODINE REMOVAL SYSTEM

### LIMITING CONDITION FOR OPERATION

3.6.2.2 The Iodine Removal System shall be OPERABLE with:

a. An spray chemical addition tank containing a level of between 90% and 100% (816 and 896 gallons) of between 33% and 35% by weight $N_2H_4$ solution, and

b. Two spray chemical addition pumps each capable of adding $N_2H_4$ solution from the spray chemical addition tank to a containment spray system pump flow.

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

ACTION:

With the Iodine Removal 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 Iodine Removal 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 Iodine Removal 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 $N_2H_4$ solution by chemical analysis.

c. By verifying that on recirculation flow, each spray chemical addition pump develops a discharge pressure of 100 psig when tested pursuant to Specification 4.0.5.

d. 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 Containment Spray Actuation (CSAS) Test Signal, and

   2. Verifying that each spray chemical addition pump starts automatically on a CSAS test signal.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 5 years by verifying each solution flow rate from the following drain connections in the Iodine Removal System:

1. (Drain line location) 0.63 ± 0.02 gpm.

2. (Drain line location) 0.63 ± 0.02 gpm.

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.6.2.3 Containment Cooling System

3/4.6.3 Iodine Cleanup System (Optional)

See Applicant's SAR

## 3/4.6.4 CONTAINMENT ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.6.4 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

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

ACTION:

1. With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE In each affected penetration that is open and either:
   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 the 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.

### SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 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.

4.6.3.2 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 its applicable actuation test signal, each isolation valve actuates to its isolation position. Valves which may be required to open or close following an accident will be actuated to demonstrate their capability to achieve both positions.

b. Verifying that on a Containment Radiation-High test signal, ** (BOP)

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*The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

**See Applicant's SAR for the containment isolation valves actuated by this signal.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

| Penetration Number | Valve Number | Function | ESF Actuation Signal | Required Post-Accident Valve Position | Maximum Valve Actuation Time (sec) |
|---|---|---|---|---|---|
| A. Remotely Actuated Valves | | | | | |
| 11 | SI-331 | Hot Leg Injection Valve | None | Open | 10 |
| 12 | SI-321 | Hot Leg Injection Valve | None | Open | 10 |
| 13<br>14<br>15<br>16 | SI-616,617<br>SI-626,627<br>SI-636,637<br>SI-646,647 | High Pressure Cold Leg Injection Valves | SIAS | Open | 10 |
| 17<br>18<br>19<br>20 | SI-615<br>SI-625<br>SI-635<br>SI-645 | Low Pressure Cold Leg Injection Valves | SIAS | Open | 10 |
| 23 | SI-673 | Containment sump isolation valve | RAS | Open | 20 |
| | SI-674 | Containment sump isolation valve | RAS | Open | 20 |
| 24 | SI-675 | Containment sump isolation valve | RAS | Open | 20 |
| | SI-676 | Containment sump isolation valve | RAS | Open | 20 |
| 27 | SI-690 | Shutdown Cooling Warmup bypass valve | None | Open or Closed | 30 |
| | SI-656 | Shutdown Cooling isolation valve | None | Open or Closed | 80 |
| | SI-654 | Shutdown Cooling isolation valve | None | Open or Closed | 80 |

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

| Penetration Number | Valve Number | Function | ESF Actuation Signal | Required Post-Accident Valve Position | Maximum Valve Actuation Time (sec) |
|---|---|---|---|---|---|
| 28 | SI-691 | Shutdown Cooling Warmup bypass valve | None | Open or Closed | 30 |
| | SI-655 | Shutdown Cooling isolation valve | None | Open or Closed | 80 |
| | SI-653 | Shutdown Cooling isolation valve | None | Open or Closed | 80 |
| 29 | SI-682 | Safety Injection Tank fill and drain isolation valve | SIAS | Closed | 5 |
| 40 | CH-523<br>CH-516 | CVCS Letdown Line Isolation Valves | CIAS<br>CIAS/SIAS | Closed<br>Closed | 5<br>5 |
| 41 | CH-524 | CVCS Charging Line Isolation Valves | None | Open or Closed | 5 |
| 43 | CH-505<br>CH-506 | Reactor Coolant Pump Controlled Bleedoff Containment Isolation Valves | CIAS<br>CIAS | Closed<br>Closed | 5<br>5 |
| 44 | CH-560<br>CH-561 | Reactor Drain Tank Suction Isolation Valves | CIAS<br>CIAS | Closed<br>Closed | 5<br>5 |
| 45 | CH-580 | Reactor Makeup Water Supply Isolation Valve to the RDT | CIAS | Closed | 5 |
| 57 | CH-255 | Seal Injection Containment Isolation Valve | None | Open or Closed | 5 |
| B. Manual Valves | | | | | |
| 29 | SI-463 | Safety Injection Tank Fill and Drain Isolation Valve | None | Closed | Not Applicable |

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

| Penetration Number | Valve Number | Function | ESF Actuation Signal | Required Post-Accident Valve Position | Maximum Valve Actuation Time (sec) |
|---|---|---|---|---|---|
| 41 | CH-393<br>CH-854 | CVCS Charging Line Isolation Valves | None<br>None | Closed<br>Closed | Not Applicable<br>Not Applicable |
| C. Check Valves | | | | | |
| 11<br>12 | SI-533<br>SI-523 | Hot Leg Injection Line Isolation Valve | None | Open | Not Applicable |
| 13<br>14<br>15<br>16 | SI-113<br>SI-123<br>SI-133<br>SI-144 | High Pressure Cold Leg Injection Line Isolation Valve | None | Open | Not Applicable |
| 17<br>18<br>19<br>20 | SI-114<br>SI-124<br>SI-134<br>SI-144 | Low Pressure Injection Line Isolation Valves | None | Open | Not Applicable |
| 41 | CH-431<br>CH-433 | CVCS Charging Line Isolation Valves | None | Open or Closed | Not Applicable |
| 45 | CH-494 | Reactor Makeup Water Supply Isolation Valve to the RDT | None | Closed | Not Applicable |
| 57 | CH-835 | Seal Injection Containment Isolation Valve | None | Open or Closed | Not Applicable |

#valves exempt from Type C testing. See Applicant's SAR.

## 3/4.6.4 COMBUSTIBLE GAS CONTROL

### HYDROGEN MONITORS

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.6.5 Combustible Gas Control

3/4.6.5.1 Hydrogen Monitors

3/4.6.5.2 Hydrogen Recombiners

3/4.6.5.3 Hydrogen Mixing System

3/4.6.6 Penetration Room Exhaust Air Cleanup System (Optional)

3/4.6.7 Vacuum Relief Valves (Optional)

3/4.6.8 Secondary Containment (Dual Type Containment, Optional)

3/4.6.8.1 Shield Building Air Cleanup System

3/4.6.8.2 Shield Building Integrity

3/4.6.8.3 Shield Building Structural Integrity

See Applicant's SAR.

## 3/4.7 PLANT SYSTEMS

### 3/4.7.1 TURBINE CYCLE

### SAFETY VALVES

#### LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

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

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam safety valves inoperable per steam generator, operation in MODES 1, 2, and 3 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the Maximum Variable Overpower trip setpoint and the Maximum Allowable Steady State Power Level are 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. Operation in MODES 3 and 4* may proceed with one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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* Until the steam generators are no longer required for heat removal.

** The maximum number of inoperable safety valves on any operating steam generator is four (4).

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOPS

| VALVE NUMBER# S/G No. 1 | S/G No. 2 | LIFT SETTING (±1%)* | MINIMUM RATED CAPACITY** |
|---|---|---|---|
| a. | | 1255 psig | 904,000 lb/hr |
| b. | | 1255 psig | 904,000 lb/hr |
| c. | | 1290 psig | 931,000 lb/hr |
| d. | | 1290 psig | 931,000 lb/hr |
| e. | | 1315 psig | 950,000 lb/hr |
| f. | | 1315 psig | 950,000 lb/hr |
| g. | | 131 psig | 950,000 lb/hr |
| h. | | 1315 psig | 950,000 lb/hr |
| i. | | 1315 psig | 950,000 lb/hr |
| j. | | 315 psig | 950,000 lb/hr |

*The lift setting pressure shall correspond to ambient conditions at the valve at nominal operating temperature and pressure.

**Capacity is rated at lift setting +3% accumulation.

#See Applicant's SAR

TABLE 3.7-2

MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL AND MAXIMUM VARIABLE OVERPOWER TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

| MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR | MAXIMUM VARIABLE OVERPOWER TRIP SETPOINT (% OF RATED THERMAL POWER) | MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL (% OF RATED THERMAL POWER) |
|---|---|---|
| 1 | 103.9 | 94.1 |
| 2 | 93.5 | 83.7 |
| 3 | 84.0 | 73.2 |
| 4 | 72.6 | 62.8 |

3/4.7.1.2 Emergency Feedwater System

See Applicant's SAR

## CONDENSATE STORAGE TANK

### LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a level of at least ** feet (300,000 gallons).

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

ACTION:

With the condensate storage tank 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)** with a water volume of at least 300,000 gallons as a backup supply to the emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The (alternate water source)** shall be demonstrated OPERABLE at least once per 12 hours whenever it is the supply source for the emergency feedwater pumps by verifying:

a. That the (alternate water source) supply line to the emergency feedwater system is open, and

b. That the (alternate water source) contains a water level of at least ** feet (300,000 gallons).

---

*Until the steam generators are no longer required for heat removed.

#Not applicable when cooldown is in progress.

**See Applicant's SAR.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 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 0.10 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.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

| TYPE OF MEASUREMENT AND ANALYSIS | SAMPLE AND ANALYSIS FREQUENCY |
|---|---|
| 1. Gross Activity Determination | At least once per 72 hours |
| 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | (a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. |
| | (b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. |

## MAIN STEAM LINE ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

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

ACTION:

MODE 1:

With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least MODE 2 within the next 6 hours.

MODES 2, 3, and 4:

With one main steam line isolation valve inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 4.6 seconds when tested pursuant to Specification 4.0.5.

4.7.1.5.2 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4 to perform the surveillance testing of Specification 4.7.1.5.1 provided the testing is performed within 12 hours after achieving normal operating steam pressure and normal operating temperature for the secondary side to perform the test.

## ATMOSPHERIC DUMP VALVES

### LIMITING CONDITION FOR OPERATION

3.7.1.6 Each atmospheric dump valve and its associated block valve shall be OPERABLE.

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

ACTION:

a. With any atmospheric dump valve(s) inoperable, restore the required atmospheric dump valve(s) to OPERABLE status within 72 hours; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN and in compliance with Specification 3.4.1.3 within the following 6 hours.

b. With any block valve(s) inoperable, restore the block valve(s) to OPERABLE status within 7 days; or be in HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours. Limit the use of associated atmospheric dump valve(s) during this period.

### SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric dump valve and its associated block valve shall be demonstrated OPERABLE:

a. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, by verifying that all valves will open and close fully.

*When steam generators are being used for decay heat removal.

## 3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

### LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than **°F when the pressure of the secondary coolant in the steam generator is greater than ** psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to ** 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 **°F.

### SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in the secondary side of the steam generators shall be determined to be less than ** psig at least once per 12 hours when the temperature of the secondary coolant is less than **°F.

**See Applicant's SAR

## 3/4.7.3 COMPONENT COOLING WATER SYSTEM

## LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.7.3 Component Cooling Water System

3/4.7.4 Service Water System

3/4.7.5 Ultimate Heat Sink

3/4.7.6 Flood Protection

3/4.7.7 Control Room Emergency Air Cleanup System

3/4.7.8 ECCS Pump Room Exhaust Air Cleanup System

See Applicant's SAR

## 3/4.7.9 SNUBBERS

### LIMITING CONDITION FOR OPERATION

3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or 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 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and 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 hydraulic and mechanical snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that type shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given type shall be performed in accordance with the following schedule:

SURVEILLANCE REQUIREMENTS (Continued)

| No. of Inoperable Snubbers of Each Type 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% |

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. 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 suspectible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.9f. When a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as-found setting, extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. Snubbers which appear inoperable during an area post maintenance inspection, area walkdown, or Transient Event Inspection shall not be considered inoperable for the purpose of establishing the Subsequent Visual Inspection Period provided that the cause of the inoperability is clearly established and remedied for that particular snubber and for the other snubbers, irrespective of type, that may be generally susceptible.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical 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 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.

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 shall be tested using one of the following sample plans. The sample plan 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 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 4.7.9f., an additional 10% 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

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 4.7.9f. 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. 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, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; 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.

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 practical 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 locations 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 testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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 4.7.9e. for snubbers not meeting the functional test acceptance criteria.

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 result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement 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 of Administrative Controls.

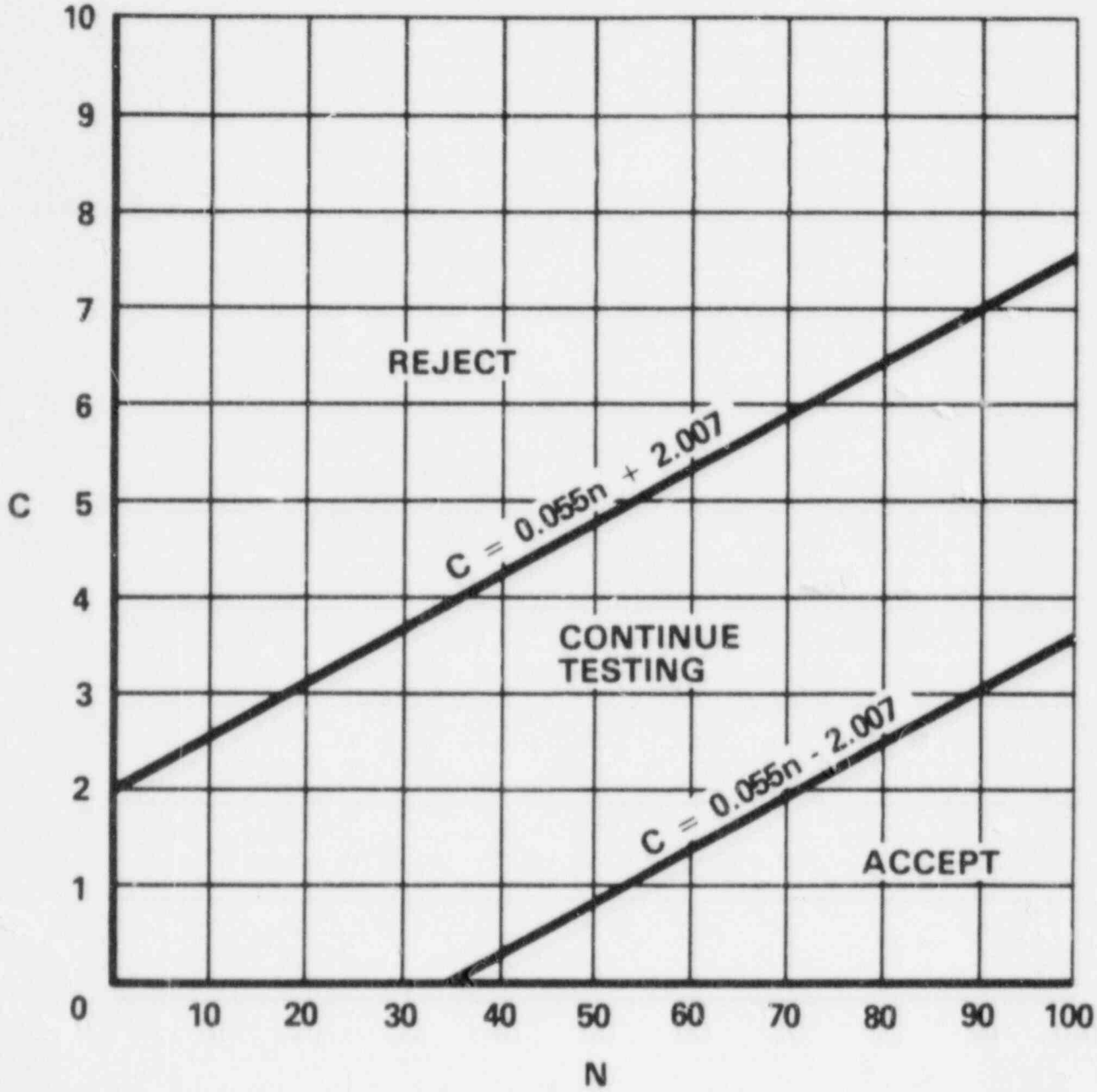


FIGURE 4.7-1

SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

## 3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS



See Applicant's SAR

## 3/4.7.14 SHUTDOWN COOLING SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.14 Two independent shutdown cooling subsystems shall be OPERABLE, with each subsystem comprised of:

a. One OPERABLE low pressure safety injection pump, and

b. An independent OPERABLE flow path capable of taking suction from the RCS hot leg and discharging coolant through the shutdown cooling heat exchanger and back to the RCS through the cold leg injection lines.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one shutdown cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within 1 hour, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and continue action to restore the required subsystem to OPERABLE status.

b. With both shutdown cooling subsystems inoperable, restore one subsystem to OPERABLE status within 1 hour or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 6 hours and continue action to restore the required subsystems to OPERABLE status.

c. With both shutdown cooling subsystems inoperable and both reactor coolant loops inoperable, initiate action to restore the required subsystems to OPERABLE status.

### SURVEILLANCE REQUIREMENTS

4.7.14 Each shutdown cooling subsystem shall be demonstrated OPERABLE:

a. At least once per 18 months, during shutdown, by establishing shutdown cooling flow from the RCS hot legs, through the shutdown cooling heat exchangers, and returning to the RCS cold legs.

b. At least once per 18 months, during shutdown, by testing the automatic and interlock action of the shutdown cooling system connections from the RCS. The shutdown cooling system suction valves shall not open when RCS pressure is greater than ** psia. The shutdown cooling system suction valves located outside containment shall close automatically when RCS pressure is greater than ** psia. The shutdown cooling system suction valve located inside containment shall close automatically when RCS pressure is greater than 700 psia.

**See applicant's SAR.

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## 3/4.8 ELECTRICAL POWER SYSTEMS

## 3/4.8.1, 3/4.8.2, 3/4.8.3 and 3/4.8.4 A.C. SOURCES, D.C. SOURCES, ONSITE POWER DISTRIBUTION SYSTEMS AND ELECTRIC EQUIPMENT PROTECTIVE DEVICES

## LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS



See Applicant's SAR

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## 3/4.9 REFUELING OPERATIONS

### 3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling pool shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a $K_{eff}$ of 0.95 or less, or

b. A boron concentration of greater than or equal to 2150 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 40 gpm of a solution containing $\geq$ 4000 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 2150 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 CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling pool shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

## 3/4.9.2 INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be OPERABLE and operating, 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. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

## 3/4.9.3 DECAY TIME

### LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

### SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

## 3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.9.4 Containment Penetrations

3/4.9.5 Communications

3/4.9.6 Refueling Machine

3/4.9.7 Crane Travel - Spent Fuel Storage Pool Building

See Applicant's SAR

## 3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

### HIGH WATER LEVEL

### LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling 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 shutdown cooling 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 shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling 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 pressure vessel hot legs.

## LOW WATER LEVEL

### LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no shutdown cooling loop 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 shutdown cooling 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 shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling 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 pressure vessel hot legs.

## 3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.9.9 Containment Purge Valve Isolation System

See Applicant's SAR

## 3/4.9.10 WATER LEVEL - REACTOR VESSEL

### FUEL ASSEMBLIES

### LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the pressure vessel.

### SURVEILLANCE REQUIREMENTS

4.9.10.1 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.

CEAs

LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the fuel seated in the reactor pressure vessel.

APPLICABILITY: During movement of CEAs within the reactor pressure vessel, when the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.2 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 CEAs.

## 3/4.9.11 WATER LEVEL - STORAGE POOL

### LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet 8 inches 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 pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

### SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage 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 pool.

## 3/4.9.12 STORAGE POOL AIR CLEANUP SYSTEM

### LIMITING CONDITION FOR OPERATION

3/4.9.12 Storage Pool Air Cleanup System

See Applicant's SAR

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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 CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2 and 3*.

ACTION:

a. With any full-length CEA 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 40 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 4000 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 and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA 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.

4.10.1.3 When in MODE 3, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs at least once per 2 hours by consideration of at least the following factors:

a. Reactor Coolant System boron concentration,

b. CEA position,

c. Reactor Coolant System average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration, and

f. Samarium concentration.

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*Operation in MODE 3 shall be limited to 6 consecutive hours.

## 3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

### LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

### SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended.

## 3/4.10.3 REACTOR COOLANT LOOPS

### LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

c. Both reactor coolant loops and at least one reactor coolant pump in each loop are in operation.

APPLICABILITY: During startup PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER or with less than the above required reactor coolant loops in operation and circulating reactor coolant, immediately trip the reactor.

### 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 startup PHYSICS TESTS.

4.10.3.2 Each logarithmic and variable overpower level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup PHYSICS TESTS.

4.10.3.3 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

## 3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

### LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient provided:

a. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

b. The condenser vacuum is maintained greater than * inches of mercury and determined as specified in Specification 4.10.4.3 below.

c. The Reactor Coolant Cold Leg temperature by narrow range indication does not exceed 568°F, determined as specified in Specification 4.10.4.4 below.

d. The Reactor Coolant Cold Leg temperature by narrow range indication is not less than 552°F, determined as specified in Specification 4.10.4.4 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1 3.1, 3.1.3.6 and 3.2.6 are suspended, either:

a. Within 15 minutes initiate corrective actions to reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1 within 1 hour, or

b. Be in HOT STANDBY within 6 hours.

With the condenser vacuum less than or equal to * inches of mercury while the requirements of Specification 3.2.6 are suspended, either:

a. Immediately reduce THERMAL POWER to satisfy the requirements of Specification 3.2.6, or

b. Trip the reactor.

With the Reactor Coolant Cold Leg temperature greater than 568°F or less than 552°F while the requirements of Specification 3.2.6 are suspended:

a. Restore the temperature to within the limits within 2 hours, or

b. Reduce the THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS (continued)

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended.

4.10.4.3 The condenser vacuum shall be determined to be greater than * inches of mercury by monitoring it at least once per 30 minutes during PHYSICS TESTS in which the requirements of Specification 3.2.6 are suspended.

4.10.4.4 The Reactor Coolant Cold Leg temperature shall be determined to be within the limits of Specifications 3.10.4.c and 3.10.4.d by monitoring the narrow range temperature indication at least once per hour during PHYSICS TESTS in which the requirements of Specification 3.2.6 are suspended.

* - See Applicant's SAR

## 3/4.10.5 MINIMUM TEMPERATURE AND PRESSURE FOR CRITICALITY

### LIMITING CONDITION FOR OPERATION

3.10.5 The minimum temperature and pressure for criticality limits of Specifications 3.1.1.4 and 3.2.8 may be suspended during low temperature PHYSICS TESTS to a minimum temperature of 300°F and a minimum pressure of 500 psia provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

b. The reactor trip setpoints on the OPERABLE Variable Overpower trip channels are set at ≤ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation required by Specification 3.4.8 except that the core critical line shown on Figure 3.4-2 does not apply.

APPLICABILITY: MODE 2*.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.8.1 prior to the next reactor criticality.

### SURVEILLANCE REQUIREMENTS

4.10.5.1 The Reactor Coolant System temperature and pressure relationship shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.

4.10.5.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour.

4.10.5.3 The Reactor Coolant System temperature shall be verified to be greater than or equal to 300°F at least once per hour.

4.10.5.4 Each Logarithmic Power Level and Variable Overpower channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

---

*First core only, prior to first exceeding 5% RATED THERMAL POWER.

## 3/4.10.6 SAFETY INJECTION TANKS

### LIMITING CONDITION FOR OPERATION

3.10.6 The safety injection tank isolation valve requirement of Specification 3.5.1a. may be suspended during partial stroke testing of the low pressure safety injection check valves (SI-114, SI-124, SI-134, SI-144) provided:

a. That power to the isolation valve is restored and the SIAS signal is not overridden.

b. Only one isolation valve at a time is closed during the testing for no longer than 1 hour.

c. That the valve is key locked opened with power removed before the next isolation valve is closed.

APPLICABILITY:

While partial stroke testing of the low pressure injection check valves during normal plant operation.

ACTION:

If the requirement of Specification 3.5.1a. was suspended to perform the Specification 3.10.6 partial stroke test and if any of the Specification 3.10.6 requirements are not met during the Specification 3.10.6 partial stroke testing, the Limiting Condition for Operation shall revert to Specification 3.5.1 and the 3.5.1 ACTION shall be applicable.

### SURVEILLANCE REQUIREMENTS

4.10.6.1 A valve alignment shall be performed within 4 hours following completion of testing to verify that all valves operated during this testing are restored to their normal positions and that power is removed from the SIT isolation valves.

## 3/4.10.7 SAFETY INJECTION TANK PRESSURE

### LIMITING CONDITION FOR OPERATION

3.10.7 The safety injection tank (SIT) pressure of Specification 3.5.1d. may be suspended for low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER;

b. The SITs have been filled per Specification 3.5.1b. and pressurized to 175 to 225 psig below the RCS pressure but not less than 254 psig;

c. All valves in the injection lines from the SITs to the RCS are open and the SITs are capable of injecting into the RCS if there is a decrease in RCS pressure.

APPLICABILITY: MODES 2, 3 and 4

ACTION:

If all the SITs do not meet the level and pressure requirements of Specification 3.10.7, restore all the SITs to meet these requirements or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.10.7.1 The THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during low pressure PHYSICS TESTS.

4.10.7.2 Every 8 hours verify:

a. All the SITs levels meet the requirements of Specification 3.5.1b.

b. All the SITs pressures meet the requirements of Specification 3.10.7.

c. The valve alignment from the SITs to the RCS has not changed.

## 3/4.11 RADIOACTIVE EFFLUENTS

## 3/4.11.1, 3/4.11.2, 3/4.11.3 and 3/4.11.4 LIQUID EFFLUENTS, GASEOUS EFFLUENTS, SOLID RADIOACTIVE WASTE AND TOTAL DOSE

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.11.1 Liquid Effluents

3/4.11.1.1 Concentration

Table 4.11-1, Radioactive Liquid Waste Sampling and Analysis Program

3/4.11.1.2 Dose

3/4.11.1.3 Liquid Holdup Tanks

3/4.11.2 Gaseous Effluents

3/4.11.2.1 Dose Rate

Table 4.11-2, Radioactive Gaseous Waste Sampling and Analysis Program

3/4.11.2.2 Dose-Noble Gases

3/4.11.2.3 Dose-Iodine-131, Iodine-133, Tritium, and Radionuclides In Particulate Form

3/4.11.2.4 Gaseous Radwaste Treatment

3/4.11.2.5 Explosive Gas Mixture

3/4.11.2.6 Gas Storage Tanks

3/4.11.3 Solid Radioactive Waste

3/4.11.4 Total Dose

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## 3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

## 3/4.12.1, 3/4.12.2 and 3/4.12.3 MONITORING PROGRAM, LAND USE CENSUS AND INTERLABORATORY COMPARISON PROGRAM

### LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.12.1 Monitoring Program

Table 3.12-1, Radiological Environmental Monitoring Program

Table 3.12-2, Reporting Levels For Radioactivity Concentrations In Environmental Samples

Table 4.12-1, Detection Capabilities For Environmental Sample Analysis

3/4.12.2 Land Use Census

3/4.12.3 Interlaboratory Comparison Program

See Applicant's SAR

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BASES

FOR

SECTION 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Section 3.0 and 4.0, but in accordance with 10 CFR 50.36, are not part of these Technical Specifications.

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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.

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 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For 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 terms of Specification 3.0.3, if both of the required containment spray systems are inoperable, within 1 hour 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 in the subsequent 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment, or components OPERABLE and (b) 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 specification 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.

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.

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.

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 thoughout 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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## 3/4.1.1 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 assuming the insertion of the regulating CEAs are within the limits of Specification 3.1.3.6, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core l fe as a function of fuel depletion, RCS boron concentration, and RCS $T_{cold}$. The most restrictive condition occurs at EOL, with $T_{cold}$ 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 6.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with the criteria used to establish the power dependent CEA insertion limits and with the assumptions used in the FSAR Safety Analysis.

With $T_{cold}$ less than or equal to 210°F, the reactivity transients resulting from uncontrolled RCS cooldown are minimal and a 4% $\Delta k/k$ SHUTDOWN MARGIN requirement is set to ensure that reactivity transients resulting from an inadvertent single CEA withdrawal event are minimal.

### 3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

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 cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

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, and (4) an emergency power supply from OPERABLE diesel generators. The nominal capacity of each charging pump is 44 gpm at its discharge. Up to 16 gpm of this may be diverted to the volume control tank via the RCP control bleedoff. Instrument inaccuracies and pump performance uncertainties are limited to 2 gpm yielding the 26 gpm value.

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 4% delta k/k after xenon decay and cooldown to 210°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires ** gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

With the RCS temperature below 210°F one injection system 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 injection system becomes inoperable. The restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

The boron capability required below 210°F is based upon providing a 4% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F to 120°F. This condition requires ** gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

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** See Applicant's SAR

BORATION SYSTEMS (Continued)

The values of water volumes, temperatures, and boron concentration in the refueling water tank are provided to ensure that the assumptions used in the initial conditions of the LOCA Safety Analysis remain valid.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

With the RCS temperature below 210°F while in MODES 5 and 6, a source of borated water is required to be available for reactivity control and makeup for losses due to contraction and evaporation. The requirement of 33,500 gallons of 4000 ppm borated water in either the refueling water tank or spent fuel pool ensures that this source is available.

3/4.1.2.7 BORON DILUTION ALARMS

The startup channel high neutron flux alarms alert the operator to an inadvertent boron dilution. Both channels must be operating to assure detection of a boron dilution event by the high neutron flux alarms. If one or both of the alarms are inoperable at any time, the bases for ACTION statements are as follows:

a. One startup channel high neutron flux alarm not operating:

With only one startup channel high neutron flux alarm OPERABLE while in MODE 3, 4, 5, or 6, a single failure to the alarm could prevent detection of boron dilution. By periodic monitoring of the RCS boron concentration by either boronometer or RCS sampling, a decrease in the boron concentration during an inadvertent boron dilution event will be observed. This provides alternate methods of detection of boron dilution with sufficient time for termination of the event before complete loss of SHUTDOWN MARGIN and return to criticality.

b. Both startup channel high neutron flux alarms not operating:

When both startup channel high neutron flux alarms are inoperable, there is no means of alarming on high neutron flux when subcritical. Therefore, either simultaneous use of the boronmeter and RCS sampling or independent collection and analysis of two RCS samples to monitor the RCS boron concentration provides alternate indications of inadvertent boron dilution. This will allow detection with sufficient time for termination of boron dilution before complete loss of shutdown margin and return to criticality.

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 CEA misalignments are limited to acceptable levels.

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.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs, and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is (1) a small effect on the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, (2) a small effect on the available SHUTDOWN MARGIN, and (3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution.

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in (1) local burnup, (2) peaking factors, and (3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO

MOVABLE CONTROL ASSEMBLIES (Continued)

and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out' limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA 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.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with $T_{cold}$ greater than or equal to 552°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.

Several design steps were employed to accommodate the possible CEA guide tube wear which could arise from CEA vibrations when fully withdrawn. Specifically, a programmed insertion schedule will be used to cycle the CEAs between the full out position ("FULL OUT" LIMIT) and 3.0 inches inserted over the fuel cycle. This cycling will distribute the possible guide tube wear over a larger area, thus minimizing any effects. To accommodate this programmed insertion schedule, the fully withdrawn position was redefined, in some cases, to be 144.75 inches (193 steps) or greater.

The establishment of LSSS and LCOs requires that the expected long- and short-term behavior of the radial peaking factors be determined. The long-term behavior relates to the variation of the steady-state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short-term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions

MOVABLE CONTROL ASSEMBLIES (Continued)

and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base loaded, or load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that 1) the minimum SHUTDOWN MARGIN is maintained, and 2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

If an inward CEA deviation event occurs, the current CPC/CEAC algorithm applies penalty factors to each of the DNB and LHR calculations. The first, a static penalty factor, is applied upon detection of the event. The second, a xenon redistribution penalty, is applied linearly as a function of time after the CEA deviation. If the penalty factors are to be removed from CPC/CEAC, in an attempt to reduce unnecessary trips, for example, the removal could be compensated for by imposing a power reduction as a function of time after the inward CEA deviation and restricting CEA insertion limits when COLSS is out of service. In addition, part length CEA maneuvering could be restricted to justify any reduction of the PLR deviation penalty factors. These restrictions should be included in the Technical Specifications preferably in the form of figures.

## 3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of 14.0 kW/ft are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady-state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the maximum linear heat rate calculated by COLSS is conservative with respect to the actual maximum linear heat rate existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB, and total core power are also monitored by the CPCs (assuming minimum core power of 20% of RATED THERMAL POWER). The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors plus those associated with the CPC startup test acceptance criteria are also included in the CPCs.

3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS ($F_{xy}^{c}$) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS ($F_{xy}^{m}$) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - $T_q$

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

The AZIMUTHAL POWER TILT is equal to $(P_{tilt}/P_{untilt})-1.0$ where:

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q\, g \cos(\theta - \theta_0)$$

where:

$T_q$ is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

$\theta$ is the azimuthal core location

$\theta_0$ is the azimuthal core location of maximum tilt

AZIMUTHAL POWER TILT - $T_q$ (Continued)

$P_{tilt}/P_{untilt}$ is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on DNBR includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limits calculated by COLSS (based on the minimum DNBR Limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux, state parameter measurement, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

The DNBR penalty factors listed in Specification 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

## 3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the safety analyses.

## 3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

## 3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of the core average AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

## 3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

## 3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation 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.

Response time testing of resistance temperature devices, which are a part of the reactor protective system, shall be performed by using in-situ loop current test techniques or another NRC approved method.

Any modifications which are made to the core protection calculator software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with an NRC approved procedure on CPC software modifications.

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If the margins are not maintained, a reactor trip will occur.

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 measurement of response time at the specified frequencies provides assurance that the protective and ESF action function 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. The response times in Table 3.3-2 are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time). The response times are taken from the sequence-of-events Tables in Section 15 of CESSAR.

REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

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 times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

See Applicant's SAR.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the 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 reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

See Applicant's SAR.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

See Applicant's SAR.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

The OPERABILITY of the remote shutdown system instrumentation ensures that sufficient capability is available to permit safe shutdown and maintenance of HOT STANDBY 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 parameters selected to be monitored ensure that (1) the condition of the reactor is known, (2) conditions in the RCS are known, (3) the steam generators are available for residual heat removal, (4) a source of water is available for makeup to the RCS, and (5) the charging system is available to makeup water to the RCS.

The OPERABILITY of the remote shutdown system insures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation and control circuits 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 shutdown the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR 50.

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-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 2, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

See Applicant's SAR for the Post-Accident Monitoring Instrumentation.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

See Applicant's SAR.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

See Applicant's SAR.

3/4.3.3.9 LOOSE-PART DETECTION INSTRUMENTATION

See Applicant's SAR.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

See Applicant's SAR.

3/4.3.4 TURBINE OVERSPEED PROTECTION

See Applicant's SAR.

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## 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.231 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 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two shutdown cooling loops be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling 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 shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 4000 gpm will circulate one equivalent Reactor Coolant System volume of 12,097 cubic feet in approximately 23 minutes. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold legs less than or equal to **°F during cooldown or **°F during heatup are provided to prevent RCS pressure transients, caused by energy additions from the secondary 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 restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

## 3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve a minimum of 460,000 lb per hour of saturated steam. 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 shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

**See Applicant's SAR.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the 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 Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

The auxiliary pressurizer spray is required to depressurize the RCS by cooling the pressurizer steam space to permit the plant to enter shutdown cooling. The auxiliary pressurizer spray is required during those periods when normal pressurizer spray is not available, such as during natural circulation and during the later stages of a normal RCS cooldown. The auxiliary pressurizer spray also distributes boron to the pressurizer when normal pressurizer spray is not available.

## 3/4.4.4 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 primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 gpm per steam generator). Cracks having a primary-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 primary-to-secondary leakage of 0.5 gpm 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 tubes with imperfections exceeding the plugging limit of ** 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.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9 of Administrative Controls prior to the 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.

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**See Applicant's SAR. Value to be determined in accordance with recommendations of Regulatory Guide 1.121, August 1976.

## 3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

### 3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

See Applicant's SAR.

### 3/4.4.5.2 OPERATIONAL LEAKAGE

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. A threshold value of less than 1 gpm is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances 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 surveillance requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 gpm for both steam generators ensures that the dosage contribution from the tube leakage will be limited to less than Part 100 guidelines for infrequent and limiting fault events. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 gpm 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.

PRESSURE BOUNDARY LEAKAGE of any magnitude may be indicative of an impending failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

## 3/4.4.6 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

**See Applicant's SAR.

3/4.4.6 CHEMISTRY (Continued)

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.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss-of-offsite electrical power. 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 ** 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 primary coolant's specific activity greater than 1.0 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 1.0 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 reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

Reducing $T_{cold}$ to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific

**See Applicant's SAR.

SPECIFIC ACTIVITY (Continued)

activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

## 3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so as not to exceed the limit lines of Figure 3.4-2. This ensures that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses at the inner wall tend to alleviate the tensile stresses induced by the internal pressure.

At the outer wall of the vessel, these thermal stresses are additive to the pressure induced tensile stresses. The magnitude of the thermal stresses at either location is dependent on the rate of heatup. Consequently, each heatup rate of interest must be analyzed on an individual basis for both the inner and outer wall.

The heatup and cooldown limit curve (Figure 3.4-2) is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curve was prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2.

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based upon the fluence and residual element content, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel

PRESSURE/TEMPERATURE LIMITS (Continued)

Materials." The heatup and cooldown limit curves Figure 3.4-2 includes predicted adjustments for this shift in $RT_{NDT}$ at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in $RT_{NDT}$ of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta $RT_{NDT}$ determined from the surveillance capsule is different from the calculated delta $RT_{NDT}$ for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimens are removed and examined to determine changes in material properties. The results of these examinations shall be used to update Figure 3.4-2 based on the greater of the following:

(1) the actual shift in reference for (Table B 3/4 4-1) as determined by impact testing, or

(2) the predicted shift in reference temperature for the limiting weld and plate as determined by RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

The maximum $RT_{NDT}$ for all Reactor Coolant System pressure-retaining materials has been determined to be $40^{o}F$. The Lowest Service Temperature limit shown on Figure 3.4-2 is based upon this $RT_{NDT}$ since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be $RT_{NDT} + 100^{o}F$ for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

(FORGINGS) and (PLATES)

| PIECE NO. | CODE NO. | MATERIAL | VESSEL LOCATION | DROP WEIGHT RESULTS (°F) | $RT_{NDT}$ (°F) | TEMPERATURE OF CHARPY V-NOTCH @ 30 ft - lb | TEMPERATURE OF CHARPY V-NOTCH @ 50 ft - lb | MINIMUM UPPER SHELF $C_v$ ENERGY FOR LONGITUDINAL DIRECTION-ft lb |
|---|---|---|---|---|---|---|---|---|
| See Applicant's SAR. | | | | | | | | |

PRESSURE/TEMPERATURE LIMITS (Continued)

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limits imposed on the RCS heatup and cooldown rates, 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 **°F during cooldown and **°F during heatup. Either one of the two SCS suction line relief valves provides 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 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

The limitations imposed on the RCS heatup and cooldown rates are provided to assure low temperature overpressure protection (LTOP) with the two shutdown cooling suction line relief valves operable. At low temperatures with the relief valves aligned to the RCS, it is necessary to restrict heatup and cooldown rates to assure that the P/T limits are not exceeded. During worst case transients, RCS peak pressures can reach the relief valve setpoint, ** psig, plus accumulation. At temperatures greater than **°F during cooldown and **°F during heatup, the heatup and cooldown rate limitations assure the limits of Appendix G to 10 CFR 50 will not be exceeded with overpressure protection provided by the primary safety valves.

3/4.4.9 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.**

**See Applicant's SAR.

## 3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head ensures the capability exists to perform this function. In addition, Branch Technical Position RSB 5-1 requires that a reactor vessel head vent path is OPERABLE in order to achieve a safe shutdown.

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 vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737.

## 3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Safety Injection System (SIS) safety injection tanks 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 safety injection tanks. This initial surge of water into the RCS provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the safety injection tanks will adequately perform their function in the event of a LOCA in MODE 1, 2, 3, or 4.

A minimum of 25% narrow range corresponding to 1790 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water are used in the safety analysis as the volume in the SITs. To allow for instrument accuracy, 28% narrow range corresponding to 1802 cubic feet and 72% narrow range corresponding to 1914 cubic feet, are specified in the Technical Specification.

A minimum of 593 psig and a maximum pressure of 632 psig at normal operating conditions are used in the safety analysis. To allow for instrument accuracy 600 psig minimum and 625 psig maximum are specified in the Technical Specification.

For MODES 3 and 4 operation with pressurizer pressure less than 1750 psia the Technical Specifications require a minimum of 57% wide range corresponding to 1361 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water per tank, when three safety injection tanks are operable and a minimum of 36% wide range corresponding to 908 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet per tank, when four safety injection tanks are operable at a minimum pressure of 235 psig and a maximum pressure of 625 psig. To allow for instrument inaccuracy, 60% wide range instrument corresponding to 1415 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when three safety injection tanks are operable, and 39% wide range instrument corresponding to 962 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when four SITs are operable, are specified in the Technical Specifications. To allow for instrument inaccuracy 254 psig is specified in the Technical Specifications.

A boron concentration of 4000 ppm minimum and 4400 ppm maximum are used in the safety analysis.

The SIT isolation valves are not single failure proof; therefore, whenever the valves are open power shall be removed from these valves and the switch keylocked open. These precautions ensure that the SITs are available during a Limiting Fault.

The SIT nitrogen vent valves are not single failure proof against depressurizing the SITs by spurious opening. Therefore, power to the valves is removed while they are closed to ensure the safety analysis assumption of four pressurized SITs.

SAFETY INJECTION TANKS (Continued)

All of the SIT nitrogen vent valves are required to be operable so that, given a single failure, all four SITs may still be vented during post-LOCA long-term cooling. Venting the SITs provides for SIT depressurization capability which ensures the timely establishment of shutdown cooling entry conditions as assumed by the safety analysis for small break LOCAs.

The limits for operation with a safety injection tank 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 safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank 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 separate and independent ECCS subsystems for MODES 1 and 2 and for MODE 3 with the pressurizer pressure greater than or equal to 1750 psia 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 safety injection tanks 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.

In MODE 3 with the pressurizer pressure less than 1750 psia and in MODE 4, 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.

The surveillance requirements provided to ensure OPERABILITY of each component ensure 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

*The following test conditions, which apply during flow balance tests, ensure that the ECCS subsystems are adequately tested.

1. The pressurizer pressure is at 15 psia.
2. The miniflow bypass recirculation lines are aligned for injection.
3. For LPSI system, (add/subtract) 3.2 gpm (to/from) the 2450 gpm requirement for every foot by which the difference of RWT water level above the RWT RAS setpoint level (exceeds/is less than) the difference of RCS water level above the cold leg centerline.

ECCS SUBSYSTEMS (Continued)

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.

The term "minimum bypass recirculation flow," as used in Specification 4.5.2e.3. and 4.5.2f., refers to that flow directed back to the RWT from the ECCS pumps for pump protection. Testing of the ECCS pumps under the condition of minimum bypass recirculation flow in Specification 4.5.2f. verifies that the performance of the ECCS pumps supports the safety analysis minimum RCS pressure assumption at zero delivery to the RCS.

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the refueling water tank (RWT) 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 RWT minimum volume and boron concentration ensure that (1) sufficient water plus 10% margin is available to permit 20 minutes of engineered safety features pump operation, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the 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.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between ** for the solution recirculated within containment after a LOCA.

The limit on the RWT solution temperature ensures that the assumptions used in the LOCA analyses remain valid.

**See Applicant's SAR.

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## 3/4.6.1 PRIMARY CONTAINMENT

See Applicant's SAR

## 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. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

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.

### 3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the Iodine Removal System ensures that sufficient $N_2H_4$ is added to the containment spray in the event of a LOCA. The limits on $N_2H_4$ volume and concentration ensure adequate chemical available to remove iodine from the containment atmosphere following a LOCA.

### 3/4.6.2.3 CONTAINMENT COOLING SYSTEM

See Applicant's SAR.

## 3/4.6.3 IODINE CLEANUP SYSTEM (OPTIONAL)

See Applicant's SAR.

## 3/4.6.4 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 GDC 54 through GDC 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 COMBUSTIBLE GAS CONTROL

See Applicant's SAR.

3/4.6.6 PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM (OPTIONAL)

See Applicant's SAR.

3/4.6.7 VACUUM RELIEF VALVES (OPTIONAL)

See Applicant's SAR.

3/4.6.8 SECONDARY CONTAINMENT (DUAL TYPE CONTAINMENT, OPTIONAL)

See Applicant's SAR.

## 3/4.7.1 TURBINE CYCLE

### 3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam safety valves (MSSVs) limit secondary system pressure to within 110% (1380 psig) of the design pressure (1255 psig) during the most severe anticipated operational transient. For design purposes the valves are sized to pass a minimum of 102% of the RATED THERMAL POWER at 102% of design pressure. The adequacy of this relieving capacity is demonstrated by maintaining the Reactor Coolant System pressure below NRC acceptance criteria (120% of design pressure for large feedwater line breaks, CEA ejection and 110% of design pressure for all overpressurization events).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code.** The total relieving capacity for all twenty MSSVs at 110% of system design pressure (adjusted for 50 psi pressure drop to valves inlet) is $18.66 \times 10^6$ lbm/hr. This capacity is less than the total rated capacity of $19 \times 10^6$ lbm/hr given in FSAR Table 5.4.13-2 as the MSSVs are operating at an inlet pressure below rated conditions. At these same secondary pressure conditions, the total steam flow at 102% (2% uncertainty) of 3817 MWt (RATED THERMAL POWER plus 17 MWt pump heat input) is $17.83 \times 10^6$ lbm/hr. The ratio of this total steam flow to the total capacity is 104.6%.

STARTUP and/or POWER OPERATION is allowable with MSSVs inoperable if the maximum allowable power level is reduced to a value equal to the product of the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator with the ratio of the total steam flow to available relieving capacity.

$$\text{Allowable Power Level} = \left(\frac{10-N}{10}\right) \times 104.6$$

The ceiling on the variable over power reactor trip is also reduced to an amount over the allowable power level equal to the BAND given for this trip in Table 2.2-1.

$$SP = \text{Allowable Power Level} + 9.8$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.

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**See Applicant's SAR for appropriate edition of the ASME Code.

SAFETY VALVES (continued)

| | | |
|---|---|---|
| 10 | = | total number of secondary safety valves for one steam generator. |
| N | = | number of inoperable main steam safety valves on the steam generator with the greater number of inoperable valves. |
| 104.6 | = | ratio of main steam safety valve relieving capacity of 110% steam generator design pressure to calculated steam flow rate at 100% plant power + 2% uncertainty (see above text) |
| 9.8 | = | BAND between the maximum thermal power and the variable overpower trip setpoint ceiling |

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

See Applicant's SAR.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank ensures that a minimum water volume of 300,000 gallons is available to maintain the Reactor Coolant System at HOT STANDBY for 8 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with concurrent total loss-of-site power, and also ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to 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 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

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.

3/4.7.1.6 ATMOSPHERIC DUMP VALVES

One atmospheric dump valve is required to hold the plant at HOT STANDBY and to conduct a controlled cooldown to shutdown cooling entry conditions; with the condenser unavailable, one steam generator unavailable for heat removal, and a single failure of one of the atmospheric dump valves on the available steam generator. The block valves are required to isolate an atmospheric dump valve which can not be tightly closed.

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 to **°F and ** psig are based on a steam generator $RT_{NDT}$ of **°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

See Applicant's SAR.

3/4.7.4 SERVICE WATER SYSTEM

See Applicant's SAR.

3/4.7.5 ULTIMATE HEAT SINK

See Applicant's SAR.

3/4.7.6 FLOOD PROTECTION

See Applicant's SAR.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

See Applicant's SAR.

3/4.7.8 ECCS PUMP ROOM AIR EXHAUST CLEANUP SYSTEM

See Applicant's SAR.

3/4.7.9 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 excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

**See Applicant's SAR.

3/4.7.9 SNUBBERS (Continued)

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 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. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original 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.

To provide assurance of snubber functional reliability one of three functional testing methods are used with the stated acceptance criteria:

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.

SNUBBERS (Continued)

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 snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.10 SEALED SOURCE CONTAMINATION

See Applicant's SAR.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

See Applicant's SAR.

3/4.7.12 FIRE-RATED ASSEMBLIES

See Applicant's SAR.

3/4.7.13 AREA TEMPERATURE MONITORING

See Applicant's SAR.

3/4.7.14 SHUTDOWN COOLING SYSTEM

The OPERABILITY of two separate and independent shutdown cooling subsystems ensures that the capability of initiating shutdown cooling in the event of an accident exists even assuming the most limiting single failure occurs. The safety analysis assumes that shutdown cooling can be initiated when conditions permit.

The limits of operation with one shutdown cooling inoperable for any reason minimize the time exposure of the plant to an accident event occurring concurrent with the failure of a component on the other shutdown cooling subsystem.

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3/4.8.1 A.C. SOURCES

See Applicant's SAR.

3/4.8.2 D.C. SOURCES

See Applicant's SAR.

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

See Applicant's SAR.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

See Applicant's SAR.

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## 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 2150 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

## 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 pressure 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 PENETRATIONS

See Applicant's SAR.

## 3/4.9.5 COMMUNICATIONS

See Applicant's SAR.

## 3/4.9.6 REFUELING MACHINE

See Applicant's SAR.

## 3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

See Applicant's SAR.

## 3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation, and circulating reactor coolant at a flow rate equal to or greater than 4000 gpm ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION (Continued)

during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) the $\Delta T$ across the core will be maintained at less than 75°F during the REFUELING MODE. The required flowrate of $\geq$ 4000 gpm ensures that ** hours after reactor shutdown sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE.

Without a shutdown cooling train in operation steam may be generated; therefore, the containment should be sealed off to prevent escape of any radioactivity, and any operations that would cause an increase in decay heat should be secured.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 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 shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

See Applicant's SAR.

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 (at least 23 feet above the top of the spent fuel in the core and above the damaged spent fuel lying on top of a spent fuel rack) is available to remove a nominal 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly for a maximum fuel rod pressurization of 1200 psig. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 STORAGE POOL AIR CLEANUP SYSTEM

See Applicant's SAR.

** See Applicant's SAR.

BASES

## 3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs 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. Although testing will be initiated from MODE 2, temporary entry into MODE 3 is necessary during some CEA worth measurements. A reasonable recovery time is available for return to MODE 2 in order to continue PHYSICS TESTING.

## 3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs 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 CEA worth, (2) determine the reactor stability index and damping factor under xenon oscillation conditions, (3) determine power distributions for non-normal CEA configurations, (4) measure rod shadowing factors, and (5) measure temperature and power coefficients. This special test exception permits MTC to exceed limits in Specification 3.1.1.3 during performance of PHYSICS TESTS.

## 3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality with less than four reactor coolant pumps in operation and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

## 3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

This special test exception permits the CEAs to be positioned beyond the insertion limits and reactor coolant cold leg temperature to be outside limits during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

## 3/4.10.5 MINIMUM TEMPERATURE AND PRESSURE FOR CRITICALITY

This special test exception permits reactor criticality at low THERMAL POWER levels with $T_{cold}$ below the minimum critical temperature and pressure during PHYSICS TESTS which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions. The Low Power Physics Testing Program at low temperature (320°F) and a pressure of 500 psia is used to perform the following tests:

1. Biological shielding survey test
2. Isothermal temperature coefficient tests
3. CEA group tests
4. Boron worth tests
5. Critical configuration boron concentration

## 3/4.10.6 SAFETY INJECTION TANKS

This special test exception permits testing the low pressure safety injection system check valves. The pressure in the injection header must be reduced below the head of the low pressure injection pump in order to get flow through the check valves. The safety injection tank (SIT) isolation valve must be closed in order to accomplish this. The SIT isolation valve is still capable of automatic operation in the event of an SIAS; therefore, system capability should not be affected.

## 3/4.10.7 SAFETY INJECTION TANK PRESSURE

This special test exception allows the performance of PHYSICS TESTS at low pressure/low temperature (600 psig, 320°F) conditions which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions.

## 3/4.11 RADIOACTIVE EFFLUENTS

## BASES

### 3/4.11.1 LIQUID EFFLUENTS

See Applicant's SAR.

### 3/4.11.2 GASEOUS EFFLUENTS

See Applicant's SAR.

### 3/4.11.3 SOLID RADIOACTIVE WASTE

See Applicant's SAR.

### 3/4.11.4 TOTAL DOSE

See Applicant's SAR.

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BASES

3/4.12.1 MONITORING PROGRAM

See Applicant's SRA.

3/4.12.2 LAND USE CENSUS

See Applicant's SAR.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

See Applicant's SAR.

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SECTION 5.0

DESIGN FEATURES

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# 5.0 DESIGN FEATURES

## 5.1 SITE

### SITE AND EXCLUSION BOUNDARIES

5.1.1 The site and exclusion boundaries 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.

## 5.2 CONTAINMENT

### CONFIGURATION

5.2.1 See Applicant's SAR.

### DESIGN PRESSURE AND TEMPERATURE

5.2.2 See Applicant's SAR.

See Applicant's SAR.

SITE AND EXCLUSION BOUNDARIES

FIGURE 5.1-1

See Applicant's SAR

LOW POPULATION ZONE
FIGURE 5.1-2

## 5.3 REACTOR CORE

### FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 241 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods or burnable poison rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of approximately 1900 grams uranium. Each burnable poison rod shall have a nominal active poison length of 136 inches. The initial core loading shall have a maximum enrichment of 3.30 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4 weight percent U-235.

### CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 76 full-length and 13 part-length control element assemblies.

## 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 of the applicable surveillance requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

### VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,900 + 300/-0 cubic feet at a nominal $T_{avg}$ of 593°F.

## 5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

## 5.6 FUEL STORAGE

### CRITICALITY

5.6.1 See Applicant's SAR.

### DRAINAGE

5.6.2 See Applicant's SAR.

### CAPACITY

5.6.3 See Applicant's SAR.

## 5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Tables 5.7-1 and 5.7-2.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| Reactor Coolant System | 500 system heatup and cooldown cycles at rates ≤ 100°F/hr. | Heatup cycle - Temperature from ≤ 70°F to ≥ 565°F; cooldown cycle - Temperature from ≥ 565°F to ≤ 70°F. |
| | 500 pressurizer heatup and cooldown cycles at rates ≤ 200°F/hr. | Heatup cycle - Pressurizer temperature from ≤ 70°F to ≥ 653°F; cooldown cycle - Pressurizer temperature from ≥ 653°F to ≤ 70°F. |
| | 10 hydrostatic testing cycles. | RCS pressurized to 3125 psia with RCS temperature between 120°F and 400°F. |
| | 480 reactor trip cycles, turbine trip cycles, and loss of reactor coolant flow. | Includes combinations of reactor trips due to operator errors, equipment malfunctions, and total loss of reactor coolant flow. |
| | 200 seismic stress cycles. | Subjection to a seismic event equal to one-half the design basis earthquake (DBE). |
| | 1 complete loss of secondary pressure cycle. | Loss of secondary pressure from either steam generator due to a complete double-ended break of a steam generator steam or feedwater nozzle. |
| | 15,000 power change cycles | Cycles from 15% to 100% full load, at a rate of 5% per minute, either increasing or decreasing. (30,000 cycles total) |
| | $10^6$ step changes of 100 psi and 10°F (20°F for surge line) | Pressure variations between the pressurizer pressure setpoint for backup heater actuation and spray valve opening. Temperature variations due to CEA controller; 2000 step change of 10% full power. |

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| | 200 primary system leak test cycles | Leak test primary system at a pressure of 2250 psia at a temperature from 120°F to 400°F. |
| Pressurizer Spray Nozzle | Calculate usage factor per Table 5.7-2. | Main spray (less than four RCP operating) with fluid $\Delta T_m$ > 200°F.<br>Auxiliary spray with fluid $\Delta T_a$ > 200°F. |

$\Delta T_m$ = The difference in temperature between the pressurizer and main spray water as adjusted by the instrument correction factor.

$\Delta T_a$ = The difference in temperature between the pressurizer and Auxiliary spray water as adjusted by the instrument correction factor.

TABLE 5.7-2

PRESSURIZER SPRAY NOZZLE USAGE FACTOR

See Applicant's SAR.

SECTION 6.0

ADMINISTRATIVE CONTROLS

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## 6.1 RESPONSIBILITY

See Applicant's SAR.

## 6.2 ORGANIZATION (includes Figures 6.2-1 and 6.2-2 and Table 6.2-1)

See Applicant's SAR.

## 6.3 UNIT STAFF QUALIFICATION

See Applicant's SAR.

## 6.4 TRAINING

See Applicant's SAR.

## 6.5 REVIEW AND AUDIT

See Applicant's SAR.

## 6.6 REPORTABLE EVENT ACTION

See Applicant's SAR.

## 6.7 SAFETY LIMIT VIOLATION

See Applicant's SAR.

## 6.8 PROCEDURES AND PROGRAMS

See Applicant's SAR.

## 6.9 REPORTING REQUIREMENTS

See Applicant's SAR.

## 6.10 RECORD RETENTION

See Applicant's SAR.

## 6.11 RADIATION PROTECTION PROGRAM

See Applicant's SAR.

## 6.12 HIGH RADIATION AREA

See Applicant's SAR.

## 6.13 PROCESS CONTROL PROGRAM

See Applicant's SAR.

## 6.14 OFFSITE DOSE CALCULATION MANUAL

See Applicant's SAR.

## 6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

See Applicant's SAR.

## 6.16 PRE-PLANNED ALTERNATE SAMPLING PROGRAM

See Applicant's SAR.

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I.D.2
PLANT SAFETY PARAMETER DISPLAY CONSOLE

SUMMARY

Each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

RESPONSE 6

The Critical Function Monitoring System (CFMS) portion of the AMS provides the primary SPDS displays in the control room, the Technical Support Center (TSC), and the Emergency Operations Facility (EOF). The Qualified Safety Parameter Display System (QSPDS) portion of the AMS provides the seismic backup SPDS.

Refer to the Accident Monitoring System (AMS) description (III.A.1.2) for more detail. Information to support plant specific implementation of the CFMS shall be provided by applicants referencing CESSAR. 11

Amendment No. 11
August 30, 1985

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II.K.1.5
IE BULLETINS ON MEASURES TO MITIGATE SMALL-BREAK LOCAs AND LOSS OF FEEDWATER ACCIDENTS—REVIEW OF ESF VALVES

SUMMARY

Review all valve positions, positioning requirements, positive controls and related test and maintenance procedures to assure proper ESF functioning.

RESPONSE

The valves of the Engineered Safety Features (ESF) Systems are designed and tested to ensure proper operation in the event of an accident. This is accomplished in several ways.

1. The valves of the ESF systems are interlocked to automatically provide the sequence of operations required after an actuation of the ESF.

2. Actuator operated valves are provided with key-operated control switches, where considered necessary, to prevent unintentional misalignment of safety injection flow paths during power operation.

3. All valves that are not required to operate on initiation of safety injection or recirculation, in the injection flow path, are locked in the post accident position. Administrative controls ensure that the valves are locked in the correct position.

4. Periodic tests and inspections are performed by the Applicant to verify proper operation of each active component of the safety injection system. This includes valves.

Amendment No. 2
March 12, 1981

II.K.2.13
THERMAL MECHANICAL REPORT--EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

SUMMARY

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

RESPONSE

As an activity for the C-E Owner's Group, Combustion Engineering has prepared CEN-189, "Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCAs with Loss of Feedwater for the Combustion Engineering NSSS," December 1981. The report was transmitted to the NRC staff by letter dated December 31, 1981 from K. P. Baskin (C-E Owners Group) to D. G. Eisenhut. 9 11

The basic CEN-189 report presents methods applicable to all C-E NSSSs, and an appendix applying these methods to plant-specific parameters was provided for each docketed C-E plant at the time of submittal (December, 1981). 9

The CFMS man/machine interface includes two color-graphic CRT's in the Control Room, two color-graphic CRT's and a line printer in the TSC and two color-graphic CRT's and a line printer in the EOF. (See Figure III.A.1.2-1) A keyboard is associated with each supplied CRT. 6

Following is a design description of the CFMS. Information to support plant specific implementation shall be provided by applicants referencing CESSAR. 11

1.0 DESIGN BASES

The CFMS design bases are divided into three areas: functional, hardware, and software.

1.1 FUNCTIONAL DESIGN BASES

A. The CFMS shall provide the capability to display the status of the following critical functions:

1. Core Reactivity Control

2. Core Heat Removal Control

3. Reactor Coolant System Inventory Control

4. Reactor Coolant System Pressure Control

5. Reactor Coolant System Heat Removal Control

6. Containment Pressure/Temperature Control

7. Containment Isolation Control 6

8. Radiation Emission Control

B. The CFMS shall alarm deviations of the critical functions.

C. The CFMS shall provide the user with concise, understandable, integrated information to assist in assessing plant status during all modes of plant operation. The CFMS displays shall utilize proven human-engineering principles.

D. The CFMS shall be capable of measuring the value of plant process input signals.

E. The CFMS shall be capable of storing the values of plant process signals for a minimum of 16 hours. The values shall be time tagged.

F. The CFMS shall be capable of determining the alarm status of each process parameter.

1. Each analog process parameter shall have the capability of 8 individual alarm settings.

   - o High-out of range alarm
   - o High-high alarm
   - o High alarm
   - o Low alarm
   - o Low-low alarm
   - o Low out of range alarm

2. Each digital process parameter shall have the capability of having one of its two-states be alarmed.

G. The CFMS shall be capable of displaying information to the operator by means of a color cathode ray tube (CRT). The CFMS shall be capable of utilizing alphanumeric data formats, shapes, symbols, color coding, and blinking for information display in accordance with established human engineering guidelines.

H. The CFMS shall be capable of utilizing greater than 20 fixed format displays (pages) for information presentation. Page selection shall be under control of the operator. Each display station (CRT and keyboard) shall be capable of independently calling up any fixed format display page in the repertoire. 6

I. The CFMS shall be capable of activating a plant annunciator.

J. The CFMS shall be capable of providing a simultaneous trend of up to four analog parameters. Analog outputs shall also be provided to allow capability for sine chart recording.

1.2 HARDWARE DESIGN BASES

A. The CFMS hardware shall have sufficient calculational and memory capacity to support the functional requirements delineated in section 2.1.

B. The CFMS hardware shall have sufficient hardware features to support the functional requirements delineated in section 2.1.

C. The CFMS input hardware shall be capable of measuring a minimum of 600 plant process signals. Analog signals shall be measured with an overall accuracy of 0.25%.

D. The CFMS hardware shall be capable of providing output to: