December 29, 1988

Docket No. 50-498

Mr. J. H. Goldberg
Group Vice-President, Nuclear
Houston Lighting & Power Company
P. O. Box 1700
Houston, Texas 77001

Dear Mr. Goldberg:

SUBJECT: ISSUANCE OF AMENDMENT NO. 4 TO FACILITY OPERATING LICENSE NPF-76 - SOUTH TEXAS PROJECT, UNIT 1

The Commission has issued the enclosed Amendment No. 4 to Facility Operating License No. NPF-76 for the South Texas Project, Unit 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated November 7, 1988.

The amendment modifies the Appendix A Technical Specifications by changing the Unit 1 TS to the Combined TS for Units 1 and 2, adding the requirement placing the positive displacement pump in a lock-out condition during modes 4, 5 and 6 to prevent cold overpressurization, adding a reactor coolant pump seal isolation header pressure interlock, and making changes to the administrative section of the TS.

A copy of the Safety Evaluation supporting the amendment is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
/s/
George F. Dick, Jr., Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 4 to NPF-76
2. Safety Evaluation

cc w/enclosures:
See next page

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CAbbate	EJordan	Plant File	CMoon

DOCUMENT NAME: STP AMENDMENT 11/28

*SEE PREVIOUS CONCURRENCES

PD4/LA*	PD4/PE*	PD4/PM	OTSB	OGC-Rockville	PD4/D
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The amendment modifies the Appendix A Technical Specifications by changing the Unit 1 TS to the Combined TS for Units 1 and 2, adding the positive displacement pump in a lock-out condition during overpressurization, adding a reactor coolant pump seal isolation header pressure interlock, and making minor changes to the administrative section of the TS.

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George F. Dick, Jr., Project Manager
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 29, 1988

Docket No. 50-498

Mr. J. H. Goldberg
Group Vice-President, Nuclear
Houston Lighting & Power Company
P. O. Box 1700
Houston, Texas 77001

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Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

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2. Safety Evaluation

cc w/enclosures:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. NPF-76

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Houston Lighting & Power Company* (HL&P) acting on behalf of itself and for the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees) dated November 7, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

*Houston Lighting & Power Company is authorized to act for the City Public Service Board of San Antonio, Central Power and Light Company and City of Austin, Texas and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 4 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

José A. Calvo

Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 29, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 4

FACILITY OPERATING LICENSE NO. NPF-76

DOCKET NO. 50-498

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove, in their entirety, the existing Unit 1 Technical Specifications and replace with the combined Technical Specifications issued with the low power license for Unit 2 on December 16, 1988; and

Remove	Insert*
xiv	xiv
xvii	xvii
xviii	xviii
3/4 0-2	3/4 0-2
3/4 0-4	3/4 0-4
3/4 3-11	3/4 3-11
3/4 3-14	3/4 3-14
3/4 3-18	3/4 3-18
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
3/4 3-24	3/4 3-24
3/4 3-26	3/4 3-26
3/4 3-29	3/4 3-29
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-34	3/4 3-34
3/4 3-36	3/4 3-36
3/4 3-38	3/4 3-38
3/4 3-40	3/4 3-40
3/4 3-42	3/4 3-42
3/4 3-44	3/4 3-44
3/4 3-45	3/4 3-45
3/4 3-47	3/4 3-47
3/4 3-49	3/4 3-49
3/4 3-55	3/4 3-55
3/4 3-56	3/4 3-56
3/4 3-57	3/4 3-57
3/4 3-58	3/4 3-58
3/4 4-4	3/4 4-4
3/4 4-38	3/4 4-38

*Some of the Technical Specification (TS) pages being replaced were those issued with Amendment Nos. 1 through 3 to the Unit 1 TS. They are being reissued at this time to reflect the issuance of the combined Units 1 and 2 TS and contain vertical lines indicating the areas of change for those amendments.

Remove	Insert*
3/4 5-3	3/4 5-3
3/4 5-8	3/4 5-8
3/4 5-11	3/4 5-11
3/4 6-3	3/4 6-3
3/4 6-10	3/4 6-10
3/4 6-18	3/4 6-18
3/4 7-1	3/4 7-1
3/4 7-10	3/4 7-10
3/4 7-12	3/4 7-12
3/4 7-14	3/4 7-14
3/4 7-31	3/4 7-31
3/4 8-11	3/4 8-11
3/4 8-14	3/4 8-14
3/4 8-16	3/4 8-16
3/4 11-4	3/4 11-4
3/4 11-8	3/4 11-8
B 3/4 0-1	B 3/4 0-1
B 3/4 0-4	B 3/4 0-4
B 3/4 0-6	B 3/4 0-6
B 3/4 3-3	B 3/4 3-3
B 3/4 4-7	B 3/4 4-7
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	B 3/4 4-12
B 3/4 4-13	B 3/4 4-13
B 3/4 4-14	B 3/4 4-14
B 3/4 4-15	B 3/4 4-15
B 3/4 5-2	B 3/4 5-2
B 3/4 11-2	B 3/4 11-2
B 3/4 11-4	B 3/4 11-4
B 3/4 12-2	B 3/4 12-2
5-4	5-4
5-5	5-5
5-6	5-6
5-7	5-7
5-8	5-8
5-9	5-9
6-1	6-1
6-3	6-3
6-4	6-4
6-5	6-5
6-7	6-7
6-10	6-10
6-12	6-12
6-14	6-14
6-23	6-23
6-24	6-24

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3/4.2.1 AXIAL FLUX DIFFERENCE	B 3/4 2-1
3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR	B 3/4 2-2
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3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

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

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

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Condition for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

LIMITING CONDITION FOR OPERATION (Continued)

3.0.5 Limiting Conditions for Operation including the associated ACTION requirements shall apply to each unit individually unless otherwise indicated as follows:

a. Whenever the Limiting Conditions for Operation refers to systems or components which are shared by both units, the ACTION requirements will apply to both units simultaneously.

b. Whenever the Limiting Conditions for Operation applies to only one unit, this will be identified in the APPLICABILITY section of the specification; and

c. Whenever certain portions of a specification contain operating parameters, Setpoints, etc., which are different for each unit, this will be identified in parentheses, footnotes or body of the requirement.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

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

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

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

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

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowed outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

4.0.6 Surveillance Requirements shall apply to each unit individually unless otherwise indicated as stated in Specification 3.0.5 for individual specifications or whenever certain portions of a specification contain surveillance parameters different for each unit, which will be identified in parentheses, footnotes or body of the requirement.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5a)	Q(17)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5a)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux (Unit 1)	S	R(4, 5a)	S/U(1), Q(9)(17)	N.A.	N.A.	2**, 3, 4, 5
Source Range, Neutron Flux (Unit 2)	S	R(4, 5b)	S/U(1), Q(9)(17)	N.A.	N.A.	2**, 3, 4, 5
7. Extended Range, Neutron Flux	S	R(4)	Q(12, 17)	N.A.	N.A.	3, 4, 5
8. Overtemperature ΔT	S	R	Q(17)	N.A.	N.A.	1, 2
9. Overpower ΔT	S	R	Q(17)	N.A.	N.A.	1, 2
10. Pressurizer Pressure--Low	S	R	Q(17)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. Pressurizer Pressure --High	S	R	Q(17)	N.A.	N.A.	1, 2
12. Pressurizer Water Level--High	S	R	Q(17)	N.A.	N.A.	1
13. Reactor Coolant Flow --Low	S	R	Q(17, 18)	N.A.	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R	Q(17,18)	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q(17)	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q(17)	N.A.	1
17. Turbine Trip						
a. Low Emergency Trip Fluid Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
18. Safety Injection Input from ESFAS	N.A.	N.A.	N.A.	R	N.A.	1, 2

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1
e. Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(15),R(16)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

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

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

(1) If not performed in previous 31 days.

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

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

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

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

(5b) With the high voltage setting varied as recommended by the manufacturer, an initial discriminator bias curve shall be measured for each detector. Subsequent discriminator bias curves shall be obtained, evaluated and compared to the initial curves.

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

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

(8) (Not Used)

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

SURVEILLANCE REQUIREMENTS

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

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train so that:

a. Each logic train is tested at least once per 36 months,

b. Each actuation train is tested at least once per 54 months*, and

c. One channel per function so that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

*If an ESFAS instrumentation channel is inoperable due to response times exceeding the limits of Table 3.3-5, perform an engineering evaluation to determine if the test failure is a result of degradation of the actuation relays. If degradation of the actuation relays is determined to be the cause, increase the ENGINEERED SAFETY FEATURES RESPONSE TIME surveillance frequency such that all trains are tested at least once per 36 months.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	19
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
c. Actuation Relays	3	2	3	1, 2, 3, 4	14
d. Containment Pressure--High-1	3	2	2	1, 2, 3, 4	15
e. Pressurizer Pressure--Low	4	2	3	1, 2, 3#	20
f. Compensated Steam Line Pressure-Low	3/steam line	2/steam line any steam line	2/steam line in each steam line	1, 2, 3#	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	19
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
c. Actuation Relays	3	2	3	1, 2, 3, 4	14
d. Containment Pressure--High-3	4	2	3	1, 2, 3	17
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	19
2) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
3) Actuation Relays	3	2	3	1, 2, 3, 4	14
4) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
b. Containment Ventilation Isolation					
1) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	18
2) Actuation Relays***	3	2	3	1, 2, 3, 4	18
3) Safety Injection***	See Item 1. above for all Safety Injection initiating functions and requirements.				
4) RCB Purge Radioactivity-High	2	1	2	1,2,3,4,5##,6##	18
5) Containment Spray-Manual Initiation	See Item 2. above for Containment Spray manual initiating functions and requirements.				
6) Phase "A" Isolation-Manual Isolation	See Item 3.a. above for Phase "A" Isolation manual initiating functions and requirements.				
c. Phase "B" Isolation					
1) Automatic Actuation Logic	2	1	2	1,2,3,4	14
2) Actuation Relays	3	2	3	1,2,3,4	14
3) Containment Pressure--High-3	4	2	3	1,2,3	17
4) Containment Spray-Manual Initiation	See Item 2. above for Containment Spray manual initiating functions and requirements.				
d. RCP Seal Injection Isolation					
1) Automatic Actuation Logic and Actuation Relays	1	1	1	1,2,3,4	16

TABL[illegible] 3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3.d. RCP Seal Injection Isolation (Continued)					
2) Charging Header Pressure - Low	1	1	1	1,2,3,4	16
Coincident with Phase "A" Isolation	See item 3.a. above for Phase "A" Isolation initiating functions and requirements				
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	2/steam line	1/steam line	2/operating steam line	1, 2, 3	24
2) System	2	1	2	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Steam Line Pressure - Negative Rate--High	3/steam line	2/steam line any steam line	2/steam line in each steam line	3###	15
d. Containment Pressure - High-2	3	2	2	1, 2, 3	15
e. Compensated Steam Line Pressure - Low	3/steam line	2/steam line any steam line	2/steam line in each steam line	1, 2, 3#	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	25
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	20
c. Deleted					
d. Deleted					
e. Safety Injection	See Item 1. for all Safety Injection initiating functions and requirements.				
f. T_{avg}-Low coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	4 (1/loop)	2	3	1, 2, 3	20

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater					
a. Manual Initiation	1/pump	1/pump	1/pump	1, 2, 3	26
b. Automatic Actuation Logic	2	1	2	1, 2, 3	22
c. Actuation Relays	3	2	3	1, 2, 3	22
d. Stm. Gen. Water Level--Low-Low Start Motor-Driven Pumps and Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any stm. gen.	3/stm. gen. in each stm. gen.	1, 2, 3	20
e. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
f. Loss of Power (Motor Driven Pumps Only)	See Item 8. below for all Loss of Power initiating functions and requirements.				
7. Automatic Switchover to Containment Sump****					
a. Automatic Actuation Logic and Actuation Relays	3-1/train	1/train	1/train	1, 2, 3, 4	19
b. RWST Level--Low-Low	3-1/train	1/train	1/train	1, 2, 3, 4	19
Coincident With: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Loss of Power					
a. 4.16 kV ESF Bus Under-voltage-Loss of Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20
b. 4.16 kV ESF Bus Under-voltage-Tolerable Degraded Voltage Coincident with SI	4/bus	2/bus	3/bus	1, 2, 3, 4	20
c. 4.16 kV ESF Bus Under-voltage - Sustained Degraded Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	21
c. Reactor Trip, P-4	2	1	2	1, 2, 3	23

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Control Room Ventilation					
a. Manual Initiation	3(1/train)	2(1/train)	3(1/train)	All	27
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Automatic Actuation Logic and Actuation Relays	3	2	3	All	27
d. Control Room Intake Air Radioactivity - High	2	1	2	All	28
e. Loss of Power	See Item 8. above for all Loss of Power initiating functions and requirements.				
11. FHB HVAC					
a. Manual Initiation	3(1/train)	2(1/train)	3(1/train)	1, 2, 3, 4 or with irradiated fuel in spent fuel pool	29, 30
b. Automatic Actuation Logic and Actuation Relays	3	2	3	1, 2, 3, 4 or with irradiated fuel in spent fuel pool	29, 30
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Spent Fuel Pool Exhaust Radioactivity - High	2	1	2	With irradiated fuel in spent fuel pool	30

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

***Function is actuated by either actuation train A or actuation train B. Actuation train C is not used for this function.

****Automatic switchover to containment sump is accomplished for each train using the corresponding RWST level transmitter.

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

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

###Trip function automatically blocked above P-11 and may be blocked below P-11 when Low Compensated Steamline Pressure Protection is not blocked.

ACTION STATEMENTS

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

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

ACTION 16 - With the Charging Header Pressure channel inoperable:

a) Place the Charging Header Pressure channel in the tripped condition within one hour and

b) Restore the Charging Header Pressure channel to operable status within 7 days or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

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

ACTION 18 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High 1	3.6	0.71	2.0	$\leq$ 3.0 psig	$\leq$ 4.0 psig
e. Pressurizer Pressure--Low	13.1	10.71	2.0	$\geq$ 1850 psig##	$\geq$ 1842 psig##
f. Compensated Steam Line Pressure-Low	13.6	10.71	2.0	$\geq$ 735 psig	$\geq$ 714.7 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High-3	3.6	0.71	2.0	$\leq$ 9.5 psig	$\leq$ 10.5 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
3) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
4) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Containment Ventilation Isolation					
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
4) RCB Purge Radioactivity-High	3.1×10^{-4} µCi/cc	1.8×10^{-4} µCi/cc	1.3×10^{-4} µCi/cc	$\leq 5 \times 10^{-4}$ µCi/cc ###	$\leq 6.4 \times 10^{-4}$ µCi/cc
5) Containment Spray - Manual Initiation	See Item 2. above for Containment Spray manual initiation Trip Setpoints and Allowable Values.				
6) Phase "A" Isolation - Manual Initiation	See Item 3.a. above for Phase "A" Isolation manual initiation Trip Setpoints and Allowable Values.				
c. Phase "B" Isolation					
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure--High-3	3.6	0.71	2.0	$\leq$ 9.5 psig	$\leq$ 10.5 psig
4) Containment Spray-Manual Initiation	See Item 2. above for Containment Spray manual initiation Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. RCP Seal Injection Isolation					
1) Automatic Acutation Logic and Activation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
2) Charging Header Pressure - Low	4.6	1.01	2.0	$\geq$ 560.0 psig	$\geq$ 495.4 psig
Coincident with Phase "A" Isolation	See Item 3.a. above for Phase "A" Isolation Setpoints and Allowable Values				
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Line Pressure - Negative Rate--High	2.6	0.5	0	$\leq$ 100 psi	$\leq$ 126.3 psi**
d. Containment Pressure - High-2	3.6	0.71	2.0	$\leq$ 3.0 psig	$\leq$ 4.0 psig
e. Compensated Steam Line Pressure - Low	13.6	10.71	2.0	$\geq$ 735 psig	$\geq$ 714.7 psig*
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--High-High (P-14)	4.5	2.35	2.0+0.2#	$\leq$ 87.5% of narrow range instrument span.	$\leq$ 88.9% of narrow range instrument span.
c. Deleted					

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (Continued)					
d. Deleted					
e. Safety Injection	See Item 1 above for all Safety Injection Trip Setpoints and Allowable Values.				
f. T_{avg}-Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	4.5	1.36	0.8	$\geq$ 574°F	$\geq$ 571.1°F
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Water Level--Low-Low	15.0	12.75	2.0+0.2#	$\geq$ 33.0% of narrow range instrument span.	$\geq$ 31.5% of narrow range instrument span.
e. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
f. Loss of Power (Motor Driven Pumps Only)	See Item 8. below for all Loss of Power Trip Setpoints and Allowable Values.				
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low Coincident With:	5.0	1.21	2.0	≥ 11%	≥ 9.1%
Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
8. Loss of Power					
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	N.A.	N.A.	N.A.	≥ 3107 volts with a ≤ 1.75 second time delay.	≥ 2979 volts with a ≤ 1.93 second time delay.
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with SI)	N.A.	N.A.	N.A.	≥ 3835 volts with a ≤ 35 second time delay.	≥ 3786 volts with a ≤ 39 second time delay.
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	N.A.	N.A.	N.A.	≥ 3835 volts with a ≤ 50 second time delay.	≥ 3786 volts with a ≤ 55 second time delay.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	$\leq$ 1985 psig	$\leq$ 1993 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$ 563°F	$\geq$ 560.1°F
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
10. Control Room Ventilation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Control Room Intake Air Radioactivity - High	3.7×10^{-5} μCi/cc	2.2×10^{-5} μCi/cc	1.6×10^{-5} μCi/cc	$\leq 6.1\times10^{-5}$ μCi/cc	$\leq 7.8\times10^{-5}$ μCi/cc
e. Loss of Power	See Item 8. above for all Loss of Power Trip Setpoints and Allowable Values.				
11. FHB HVAC					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. FHB HVAC (Continued)					
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
d. Spent Fuel Pool Exhuast Radioactivity - High	3.1×10^{-4} µCi/cc	1.8×10^{-4} µCi/cc	1.3×10^{-4} µCi/cc	$\leq 5.0 \times 10^{-4}$ µCi/cc	$\leq 6.4 \times 10^{-4}$ µCi/cc

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

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

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

#2.0% span for Steam Generator Level; 0.2% span for Reference Leg RTDs

##Until resolution of the Veritrak transmitter uncertainty issue, the trip setpoint will be set at ≥ 1869 psig, with the allowable value at ≥ 1861 psig.

###This setpoint value may be increased up to the equivalent limits of Specification 3.11.2.1 in accordance with the methodology and parameters of the ODCM during containment purge or vent for pressure control, ALARA and respirable air quality considerations for personnel entry.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Ventilation Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Cooling Water	N.A.
j. Reactor Containment Fan Coolers	N.A.
k. Control Room Ventilation	N.A.
l. Reactor Trip	N.A.
m. Start Diesel Generator	N.A.
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	$\leq 2^{(3)}$
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Phase "A" Isolation	$\leq 33^{(1)}/23^{(2)}$
4) Containment Ventilation Isolation (18-inch lines)	$\leq 23^{(1)}/13^{(2)}$
5) Auxiliary Feedwater	≤ 60
6) Essential Cooling Water	$\leq 62^{(1)}/52^{(2)}$
7) Reactor Containment Fan Coolers	$\leq 38^{(1)}/28^{(2)}$
8) Control Room Ventilation	$\leq 72^{(1)}/62^{(2)}$
9) Start Standby Diesel Generators	≤ 12

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	$\leq 2^{(3)}$
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Phase "A" Isolation	$\leq 33^{(1)}/23^{(2)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	≤ 60
6) Essential Cooling Water	$\leq 62^{(1)}/52^{(2)}$
7) Reactor Containment Fan Coolers	$\leq 38^{(1)}/28^{(2)}$
8) Control Room Ventilation	$\leq 72^{(1)}/62^{(2)}$
9) Start Standby Diesel Generators	≤ 12
4. Deleted	
5. Compensated Steam Line Pressure--Low	
a. Safety Injection (ECCS)	$\leq 22^{(4)}/12^{(5)}$
1) Reactor Trip	$\leq 2^{(3)}$
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Phase "A" Isolation	$\leq 33^{(1)}/23^{(2)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	≤ 60
6) Essential Cooling Water	$\leq 62^{(1)}/52^{(2)}$
7) Reactor Containment Fan Coolers	$\leq 38^{(1)}/28^{(2)}$
8) Control Room Ventilation	$\leq 72^{(1)}/62^{(2)}$
9) Start Diesel Generators	≤ 12
b. Steam Line Isolation	$\leq 8^{(3)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Containment Pressure--High-3	
a. Containment Spray	$\leq 30^{(1)}/20^{(2)}$
b. Phase "B" Isolation	$\leq 28^{(1)}/18^{(2)}$
7. Containment Pressure--High-2	
Steam Line Isolation	$\leq 7^{(3)}$
8. Steam Line Pressure - Negative Rate--High	
Steam Line Isolation	N.A.
9. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq 3^{(3)}$
b. Feedwater Isolation	$\leq 12^{(3)}$
10. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-Driven Auxiliary Feedwater Pump	≤ 60
11. RWST Level--Low-Low Coincident with Safety Injection	
Automatic Switchover to Containment Sump	$\leq 32^{(2)}$
12. Loss of Power	
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	≤ 12
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with Safety Injection)	≤ 49
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	≤ 65

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
13. RCB Purge Radioactivity-High	
a. Containment Ventilation Isolation (48-inch lines)	$\leq 73^{(2)}$
b. Containment Ventilation Isolation (18-inch lines)	$\leq 23^{(2)}$
14. Deleted	
15. Deleted	
16. T_{avg} - Low Coincident with Reactor Trip	
Feedwater Isolation	N.A.
17. Control Room Intake Air Radioactivity - High	
Control Room Ventilation	$\leq 78^{(2)}$
18. Spent Fuel Pool Exhaust Radioactivity - High	
FHB HVAC Emergency Startup	$\leq 42^{(2)}$
19. Charging Header Pressure - Low	N.A.

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

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

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

(3) Not dependent upon diesel generator starting or sequence loading delays.

(4) Diesel generator starting and sequence loading delay included. Low Head Safety Injection pumps not included.

(5) Diesel generator starting delays not included, sequence loading delay is included. Low Head Safety Injection pumps not included.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	M(6)	Q(4,5)	1, 2, 3, 4
d. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
e. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Compensated Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	M(6)	Q	1, 2, 3, 4
d. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4
3) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	M(6)	Q(4)	1, 2, 3, 4
4) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Containment Ventilation Isolation								
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	M(6)	Q	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) RCB Purge Radioactivity-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1,2,3,4,5*
5) Containment Spray - Manual Initiation	See Item 2. above for Containment Spray manual initiation Surveillance Requirements.							
6) Phase "A" Isolation-Manual Initiation	See Item 3.a. above for Phase "A" Isolation manual initiation Surveillance Requirements.							
c. Phase "B" Isolation								
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1,2,3,4
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	M(6)	Q	1,2,3,4
3) Containment Pressure--High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1,2,3
4) Containment Spray-Manual Initiation	See Item 2. above for Containment Spray manual initiation Surveillance Requirements.							
d. RCP Seal Injection Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A	N.A.	N.A.	N.A.	N.A.	Q	Q	1,2,3,4
2) Charging Header Pressure - Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1,2,3,4
Coincident with Phase "A" Isolation	See Item 3.a. above for Phase "A" surveillance requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(6)	Q	1, 2, 3
c. Steam Line Pressure-Negative Rate-High	S	R	M	N.A.	N.A.	N.A.	N.A.	3
d. Containment Pressure - High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Compensated Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(6)	Q(4)	1, 2, 3
b. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Deleted								
d. Deleted								
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Turbine Trip and Feedwater Isolation (Continued)								
f. T_{avg}-Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	M(6)	Q	1, 2, 3
d. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Loss of Power	See Item 8. below for all Loss of Power Surveillance Requirements.							
7. Automatic Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(6)	M(6)	Q	1, 2, 3, 4
b. RWST Level--Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A	1, 2, 3, 4
Coincident With: Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Loss of Power								
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with SI)	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
10. Control Room Ventilation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Control Room Ventilation (Continued)								
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(6)	N.A.	N.A.	All
d. Control Room Intake Air Radioactivity-High	S	R	M	N.A.	N.A.	N.A.	N.A.	All
e. Loss of Power	See Items 8. above for all Loss of Power Surveillance Requirements.							
11. FHB HVAC								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4, or with irradiated fuel in the spent fuel pool
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(6)	N.A.	N.A.	1, 2, 3, 4, or with irradiated fuel in the spent fuel pool

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. FHB HVAC (Continued)								
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
d. Spent Fuel Pool Exhaust Radio-activity-High	S	R	M	N.A.	N.A.	N.A.	N.A.	With irradiated fuel in spent fuel pool.

TABLE NOTATION

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

(2) Deleted

(3) Deleted

(4) Except relays K807, K814, K829 (Train B only), K831, K845, K852 and K854 (Trains B and C only) which shall be tested at least once per 18 months during refueling and during each COLD SHUTDOWN exceeding 24 hours unless they have been tested within the previous 92 days.

(5) Except relay K815 which shall be tested at indicated interval only when reactor coolant pressure is above 700 psig.

(6) Each actuation train shall be tested at least every 92 days on a STAGGERED TEST BASIS. Testing of each actuation train shall include master relay testing of both logic trains. If an ESFAS instrumentation channel is inoperable due to failure of the Actuation Logic Test and/or Master Relay Test, increase the surveillance frequency such that each train is tested at least every 62 days on a STAGGERED TEST BASIS unless the failure can be determined by performance of an engineering evaluation to be a single random failure.

*During CORE ALTERATIONS or movement of irradiated fuel within containment.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

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

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

TABLE 3.3-7
SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS (Unit 1 only)	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerometers***		
a. Free Field	±3g	1
b. Containment Bldg. Foundation (Tendon Gallery El. -36'9")	±3g	1
c. Outside Face Containment Shell (Reactor Containment Building El. 68'0")	±3g	1
d. Steam Generator Upper Lateral Support (Reactor Containment Building El. 66'7½")	±3g	1
e. Fuel Handling Building Foundation (Fuel Handling Building El. -29'0")	±3g	1
f. Mechanical Electrical Auxiliary Building (Mechanical Electrical Auxiliary Building El. 35'0")	±3g	1
2. Triaxial Peak Accelerographs		
a. Spent Fuel Pool Heat Exchanger (Inlet Line Fuel Handling Building El. 64'5¼")	±3g	1
b. Reactor Vessel (Reactor Containment Building El. 68'0")	±3g	1
c. Cold Leg of RC Piping (Reactor Containment Building El. 34'3")	±3g	1
3. Self-Contained Triaxial Accelerograph (At Reactor Containment Building Foundation Tendon Gallery El. -36'9")	±3g	1
4. Triaxial Seismic Switch* ** #	0.03 to 3g	1
5. Triaxial Seismic Trigger* ** ##	0.003 to 0.3g	1
6. Response Spectrum Analyzer* **	1 to 32 Hz	1
7. Magnetic Tape Recorders**	0.1 to 33 Hz	6
8. Playback System**	N.A.	1

*With reactor control room indication and alarm in Unit 1 (Alarm only in Unit 2)
**At seismic monitoring panel in Control Room, Unit 1
***Accelerometer data is gathered and analyzed by the Response Spectrum Analyzer (Item 6).
#Triaxial seismic switch is set at the OBE acceleration level of 0.05g horizontal and 0.033g vertical.
##Triaxial seismic trigger is set at 0.02g all axes.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS (Unit 1 only)	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Time-History Accelerometers***			
a. Free Field	M	R	SA
b. Containment Bldg. Foundation (Tendon Gallery El. -36'9")	M	R	SA
c. Outside Face Containment Shell (Reactor Containment Building El. 68'0")	M	R	SA
d. Steam Generator Upper Lateral Support (Reactor Containment Building El. 66'7½")	M	R	SA
e. Fuel Handling Building Foundation (Fuel Handling Building El. -29'0")	M	R	SA
f. Mechanical Electrical Auxiliary Building (Mechanical Electrical Auxiliary Building El. 35'0")	M	R	SA
2. Triaxial Peak Accelerographs			
a. Spent Fuel Pool Heat Exchanger (Inlet Line Fuel Handling Building El. 64'5¼")	N.A.	R	N.A.
b. Reactor Vessel (Reactor Containment Building El. 68'0")	N.A.	R	N.A.
c. Cold Leg of RC Piping (Reactor Containment Building El. 34'3")	N.A.	R	N.A.
3. Self-Contained Triaxial Accelerograph (At Reactor Containment Building Foundation Tendon Gallery El. -36'9")	M	R	SA
4. Triaxial Seismic Switch* **	M	R	SA
5. Triaxial Seismic Trigger* **	M	R	SA
6. Response Spectrum Analyzer* **	M	R	SA
7. Magnetic Tape Recorders**	M	R	SA
8. Playback System**	M	R	N.A.

*With reactor control room indication and alarm in Unit 1 (Alarm only in Unit 2)

**At seismic monitoring panel in Control Room, Unit 1

***Accelerometer data is gathered and analyzed by the Response Spectrum Analyzer (Item 6).

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,**

e. RHR Loop A with valve CV0198 locked or pinned in position to limit flow to 125 gpm,

f. RHR Loop B with valve CV0198 locked or pinned in position to limit flow to 125 gpm, and

g. RHR Loop C with valve CV0198 locked or pinned in position to limit flow to 125 gpm.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

---

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s) and/or RHR pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% narrow range at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant loop, or one RHR loop with valve CV0198 locked or pinned in position to limit flow to 125 gpm shall be verified in operation and circulating reactor coolant at least once per 12 hours.

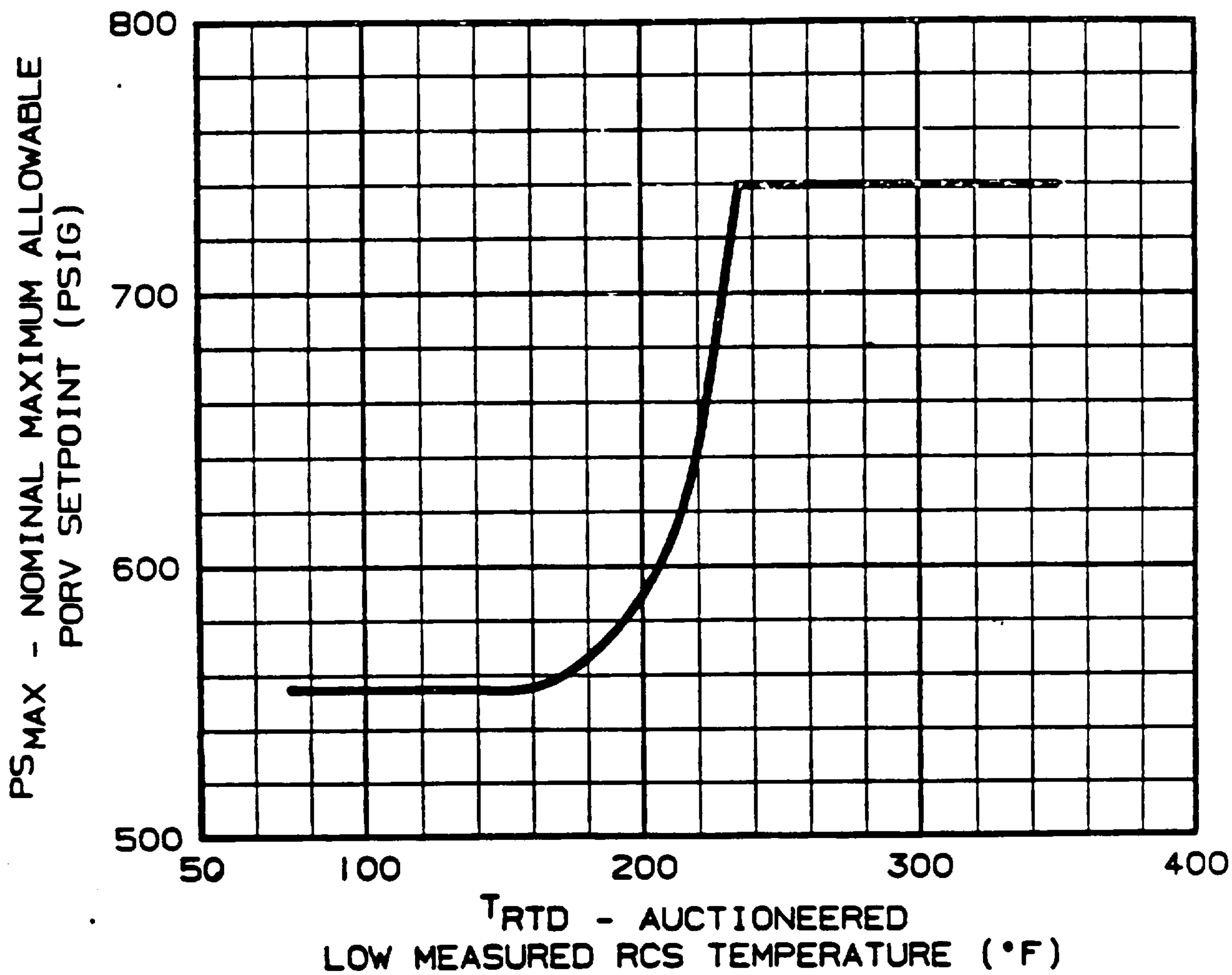

FIGURE 3.4-4

NOMINAL MAXIMUM ALLOWABLE PORV SETPOINT FOR THE COLD OVERPRESSURE SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV block valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

4.4.9.3.3 The positive displacement pump shall be demonstrated inoperable** at least once per 31 days, except when the reactor vessel head is removed or when both centrifugal charging pumps are inoperable and secured, by verifying that the motor circuit breakers are secured in the open position.***

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

**The provisions of 3.0.4 and 4.0.4 are not applicable for entry into MODE 4 from MODE 3 for the positive displacement pump declared inoperable pursuant to Specification 4.4.9.3.3 provided that the positive displacement pump is declared INOPERABLE within 4 hours after entry into MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first.

***The positive displacement pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Three independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE High Head Safety Injection pump,

b. One OPERABLE Low Head Safety Injection pump

c. One OPERABLE RHR heat exchanger, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation through a High Head Safety Injection pump and into the Reactor Coolant System and through a Low Head Safety Injection pump and its respective RHR heat exchanger into the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With less than the above subsystems OPERABLE, but with at least two High Head Safety Injection pumps in an OPERABLE status, two Low Head Safety Injection pumps and associated RHR heat exchangers in an OPERABLE status, and sufficient flow paths to accommodate these OPERABLE Safety Injection pumps and RHR heat exchangers, restore the inoperable subsystem(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

---

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection pumps declared inoperable pursuant to Specification 4.5.3.1.2 provided that the Safety Injection pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
XSI0008 A,B,C	High Head Hot Leg Recirculation Isolation	Closed
XRH0019 A,B,C	Low Head Hot Leg Recirculation Isolation	Closed

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

SURVEILLANCE REQUIREMENTS

4.5.3.1.1 The ECCS components shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.1.2 All High Head Safety Injection pumps, except the above allowed OPERABLE pumps, shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first, and at least once per 31 days thereafter.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.5.3.2 All High Head Safety Injection pumps shall be inoperable.

APPLICABILITY: MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all High Head Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.3.2 All High Head Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position at least once per 31 days.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

3/4.5.6 RESIDUAL HEAT REMOVAL (RHR) SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.6 Three independent Residual Heat Removal (RHR) loops shall be OPERABLE with each loop comprised of:

a. One OPERABLE RHR pump,

b. One OPERABLE RHR heat exchanger, and

c. One OPERABLE flowpath capable of taking suction from its associated RCS hot leg and discharging to its associated RCS cold leg.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one RHR loop inoperable, restore the required loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two RHR loops inoperable, restore at least two RHR loops to OPERABLE status within 24 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three RHR loops inoperable, immediately initiate corrective action to restore at least one RHR loop to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.5.6.1 Each RHR loop shall be demonstrated OPERABLE pursuant to the requirements of Specification 4.0.5.

4.5.6.2 At least once per 18 months by verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System to ensure that:

a. With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 350 psig, the interlocks prevent the valves from being opened, and

b. With a simulated or actual Reactor Coolant System pressure signal less than or equal to 700 psig, the interlocks will cause the valves to automatically close.

*Valves MOV-0060 A, B, and C and MOV-0061 A, B, and C may have power removed to support the FHAR (Fire Hazard Analysis Report) assumptions.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, is in accordance with the following equation:

$$|L_c-(L_{am}+L_o)| \leq 0.25\ L_a$$

where L_{am} is the measured Type A test leakage and L_o is the superimposed leak;

2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 37.5 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks,

2) Purge supply and exhaust isolation valves with resilient material seals, and

3) Penetrations using continuous Leakage Monitoring Systems.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2 or 4.6.1.7.3, as applicable;

SURVEILLANCE REQUIREMENTS (Continued)

g. Leakage from isolation valves that are sealed with fluid from a Seal System may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the Seal System and valves are pressurized to at least 1.10 P_a, 41.25 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days;

h. Type B tests for penetrations employing a continuous Leakage Monitoring System shall be conducted at P_a, 37.5 psig, at intervals no greater than once per 3 years; and

i. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

a. With more than one tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specification 4.6.1.6, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Tendons. The containment tendons' structural integrity shall be demonstrated at the end of 1, 3, and 5 years following the initial containment structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a random but representative sample of at least 13 tendons (4 inverted U and 9 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection, one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon, should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and accceptable. Unless there is abnormal degradation of the containment during the first three inspections, the sample population for subsequent inspections shall include at least 5 tendons (2 inverted U and 3 hoop);

SURVEILLANCE REQUIREMENTS (Continued)

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (inverted U and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire that:

1) The tendon wires are free of corrosion, cracks, and damage,

2) There are not changes in the presence or physical appearance of the sheathing filler grease, and

3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at 20%, 60%, and 100% of the maximum jacking force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off stresses exceed the average minimum design value given below, which are adjusted to account for elastic and time dependent losses; and

Inverted U		126 ksi
Hoop:	Cylinder	128 ksi
	Dome	123 ksi

e. Verifying the OPERABILITY of the sheathing filler grease by:

1) No voids in excess of 5% of the net duct volume,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Three independent groups of Reactor Containment Fan Coolers (RCFC) shall be OPERABLE with a minimum of two units in two groups and one unit in the third group.

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

ACTION:

With one group of the above required Reactor Containment Fan Coolers inoperable, restore the inoperable group of RCFC to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of Reactor Containment Fan Coolers shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1) Starting each non-operating fan group from the control room, and verifying that each fan group operates for at least 15 minutes, and

   2) Verifying a cooling water flow rate of greater than or equal to 550 gpm to each cooler.

b. At least once per 18 months by verifying that each fan group starts automatically on a Safety Injection test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE with isolation times less than or equal to the required isolation times.

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

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves 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 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Containment Ventilation Isolation test signal, each purge and exhaust valve actuates to its isolation position; and

c. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position.

d. Verifying that on a Phase "A" Isolation test signal, coincident with a low charging header pressure signal, that each seal injection valve actuates to its isolation position.

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

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 There are no additional requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	87
2	65
3	43

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

ATMOSPHERIC STEAM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 At least four atmospheric steam relief valves and associated manual controls shall be OPERABLE.

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

ACTION:

a. With one less than the required atmospheric steam relief valves OPERABLE, restore the required atmospheric steam relief valves to OPERABLE status within 7 days; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours and place the required RCS/RHR loops in operation for decay heat removal.

b. With two less than the required atmospheric relief valves OPERABLE, restore at least three atmospheric relief 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 and place the required RCS/RHR loops in operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric relief valve and associated manual controls shall be demonstrated OPERABLE 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 by operations of manual controls.

*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 temperatures of both the reactor and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least three independent component cooling water loops shall be OPERABLE.

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

ACTION:

With only two component cooling water loops OPERABLE, restore at least three loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least three component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve outside containment (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment or isolating the non-nuclear safety portion of the system actuates to its correct position on a Safety Injection, Loss of Offsite Power, Containment Phase "B" Isolation, or Low Surge Tank test signal, as applicable,

   2) Each Component Cooling Water System pump starts automatically on a Safety Injection or Loss of Offsite Power test signal, and

   3) The surge tank level instrumentation which provides automatic isolation of portions of the system is demonstrated OPERABLE by performance of a CHANNEL CALIBRATION test.

c. By verifying that each valve inside containment (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position prior to entering MODE 4 following each COLD SHUTDOWN of greater than 72 hours if not performed within the previous 31 days.

3/4.7.4 ESSENTIAL COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least three independent essential cooling water loops shall be OPERABLE.

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

ACTION:

With only two essential cooling water loops OPERABLE, restore at least three loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least three essential cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position;

b. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection, ECW pump start, screen wash booster pump start and essential chiller start test signals, as applicable,

2) Each Essential Cooling Water pump starts automatically on a Safety Injection or a Loss of Offsite Power test signal, and

3) Each screen wash booster pump and the traveling screen start automatically on a Safety Injection test signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation 25.5 feet Mean Sea Level, USGS datum, and

b. An Essential Cooling Water intake temperature of less than or equal to 99°F.

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

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION is applicable to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the intake water temperature and water level to be within their limits.

3/4.7.13 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.13 The temperature of each area shown in Table 3.7-3 shall not be exceeded for more than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With the temperature inside any QDPS auxiliary processing cabinet exceeding 110°F for more than 12 hours, prepare an engineering evaluation within the next 24 hours to determine the temperature effects on QDPS OPERABILITY and service life. The provisions of Specification 3.0.3 are not applicable.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-3 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specification 3.0.3 are not applicable.

c. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-3 by more than 30°F, prepare and submit a Special Report as required by ACTION b. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.13 The temperature in each of the areas shown in Table 3.7-3 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-3

AREA TEMPERATURE MONITORING

	AREA	TEMPERATURE LIMIT (°F)
1.	Relay Room (Electrical Auxiliary Building El. 35'0")	≤ 78
2.	Switchgear Rooms (Electrical Auxiliary Building El. 10'0", 35'0", 60'0")	≤ 85
3.	Electrical Penetration Spaces (Electrical Auxiliary Building El. 10'0", 35'0", 60'0")	≤ 103
4.	Safety Injection and Containment Spray Pump Cubicles(Fuel Handling Building El. -29'0")	≤ 101
5.	Component Cooling Water Pump Cubicles (Mechanical Auxiliary Building El. 10'0")	≤ 112
6.	Centrifugal Charging Pump Cubicles (Mechanical Auxiliary Building El. 10'0")	≤ 132
7.	Hydrogen Analyzer Room (Mechanical Auxiliary Building El. 60'0")	≤ 102
8.	Boric Acid Transfer Pump Cubicles (Mechanical Auxiliary Building El. 10'0")	≤ 101
9.	Standby Diesel Generator Rooms (Diesel Generator Building El. 25'0")	≤ 101*
10.	Essential Cooling Water Pump Rooms (Essential Cooling Water Intake Structure El. 34'0")	≤ 101
11.	Isolation Valve Cubicles (Isolation Valve Cubicle El. 10' 0")	≤ 101
12.	Qualified Display Processing System Rooms (Electrical Auxiliary Building El. 10'0")	≤ 94**

*Temperature limit is ≤ 120°F when testing the standby diesel generator pursuant to Surveillance Requirement 4.8.1.1.2.e.7).

**Measurement inside QDPS auxiliary processing cabinets.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 135 volts, by verifying that:

   1) The parameters in Table 4.8-2 meet the Category 3 limits,

   2) There is no visible corrosion at either cell-to-cell or terminal connections, or the connection resistance of these items is less than or equal to 150×10^{-6} ohm, and

   3) The average electrolyte temperature of six connected cells is above 65° F.

c. At least once per 18 months by verifying that:

   1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

   2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

   3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm, and

   4) The battery charger will supply at least 300 amperes at 125 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated ESF loads for the design duty cycle when the battery is subjected to a battery service test;

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A[1]	CATEGORY B[2]	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE[3] VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and < 1/4" above maximum level indication mark	>Minimum level indication mark, and < 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[6]	> 2.07 volts
Specific Gravity[4]	≥ 1.200[5]	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195[5]

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided that Category B parameters(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, Channel I 125-volt Battery Bank E1A11 (Unit 1), E2A11 (Unit 2), and Channel IV 125-volt battery bank E1C11 (Unit 1), E2C11 (Unit 2), and their two associated chargers shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required battery banks and/or charger(s) inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required battery banks and/or chargers to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a 2.0 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

ELECTRICAL POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner:

a. Train A A.C. ESF Busses consisting of:
   1) 4160-Volt ESF Bus # E1A (Unit 1), E2A (Unit 2), and
   2) 480-Volt ESF Busses # E1A1 and E1A2 (Unit 1), E2A1 and E2A2 (Unit 2) from respective load center transformers.

b. Train B A.C. ESF Busses consisting of:
   1) 4160-Volt ESF Bus # E1B (Unit 1), E2B (Unit 2), and
   2) 480-Volt ESF Busses # E1B1 and E1B2 (Unit 1), E2B1 and E2B2 (Unit 2) from respective load center transformers.

c. Train C A.C. ESF Busses consisting of:
   1) 4160-Volt ESF Bus # E1C (Unit 1), E2C (Unit 2), and
   2) 480-Volt ESF Busses # E1C1 and E1C2 (Unit 1), E2C1 and E2C2 (Unit 2) from respective load center transformers.

d. 120-Volt A.C. Vital Distribution Panels DP1201 and DP001 energized from their associated inverters connected to D.C. Bus # E1A11* (Unit 1), E2A11* (Unit 2),

e. 120-Volt A.C. Vital Distribution Panel DP1202 energized from its associated inverter connected to D.C. Bus # E1D11* (Unit 1), E2D11* (Unit 2),

f. 120-Volt A.C. Vital Distribution Panel DP1203 energized from its associated inverter connected to D.C. Bus # E1B11* (Unit 1), E2B11* (Unit 2),

g. 120-Volt A.C. Vital Distribution Panels DP1204 and DP002 energized from their associated inverters connected to D.C. Bus # E1C11* (Unit 1), E2C11* (Unit 2),

h. 125-Volt D.C. Bus E1A11 (Unit 1) E2A11 (Unit 2) energized from Battery Bank E1A11 (Unit 1), E2A11 (Unit 2),

i. 125-Volt D.C. Bus E1D11 (Unit 1) E2D11 (Unit 2) energized from Battery Bank E1D11 (Unit 1), E2D11 (Unit 2),

j. 125-Volt D.C. Bus E1B11 (Unit 1) E2B11 (Unit 2) energized from Battery Bank E1B11 (Unit 1), E2B11 (Unit 2), and

k. 125-Volt D.C. Bus E1C11 (Unit 1) E2C11 (Unit 2) energized from Battery Bank E1C11 (Unit 1), E2C11 (Unit 2).

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*The inverter(s) associated with one channel may be disconnected from its D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on its associated battery bank provided: (1) its vital distribution panels are energized, and (2) the vital distribution panels associated with the other battery banks are energized from their associated inverters and connected to their associated D.C. busses.

LIMITING CONDITION FOR OPERATION (Continued)

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

ACTION:

a. With one of the required trains of A.C. ESF busses not fully energized, reenergize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital distribution panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. distribution panel within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital distribution panel from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. Train A and Train C of A.C. ESF busses E1A and E1C (Unit 1), E2A and E2C (Unit 2), each consisting of one 4160-volt ESF bus and two 480-volt A.C. ESF load centers,

b. Four 120-volt A.C. vital distribution panels consisting of DP001, DP1201, DP002, and DP1204 energized from their associated inverter connected to its respective D.C. bus E1A11 and E1C11 (Unit 1), E2A11 and E2C11 (Unit 2), and

c. Channel I and Channel IV 125-volt D.C. busses energized from their associated battery banks E1A11 and E1C11 (Unit 1), E2A11 and E2C11 (Unit 2).

APPLICABILITY MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible, and within 8 hours, depressurize and vent the RCS through at least a 2.0 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

LIQUID WASTE PROCESSING SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Waste Processing System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-4) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Waste Processing System not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Waste Processing Systems are not being fully utilized.

4.11.1.3.2 The installed Liquid Waste Processing System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each unprotected outdoor tank shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any unprotected outdoor tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each unprotected outdoor tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

*Tanks included in this specification are those outdoor tanks that are either not surrounded by liners, dikes, or walls capable of holding the tank contents or that do not have tank overflows and surrounding area drains connected to the Liquid Waste Processing System.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit the the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS WASTE PROCESSING SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS WASTE PROCESSING SYSTEM shall be OPERABLE and appropriate portions of this system shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when the GASEOUS WASTE PROCESSING SYSTEM is not being fully utilized.

4.11.2.4.2 The installed GASEOUS WASTE PROCESSING SYSTEM shall be considered OPERABLE by meeting Specifications 3.11.2.1, and either 3.11.2.2 or 3.11.2.3.

3.4.0 APPLICABILITY

BASES

Specification 3.0.1 through 3.0.5 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action. permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. it is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into a MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.4 establishes limitations on MODE changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in MODES were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation are not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

Specification 3.0.5 delineates the applicability of each specification to Unit 1 and Unit 2 operation.

Specifications 4.0.1 through 4.0.6 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the conditions under which the specified time interval for Surveillance Requirements may be extended. Item a. permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. Item b. limits the use of the provisions of item a. to ensure that it is not used repeatedly to extend the surveillance interval beyond that specified. The limits of Specification 4.0.2 are based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. These provisions are sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the

OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with ACTION requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.5 establishes the requirement 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 shall 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. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the 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. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, 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.

Specification 4.0.6 delineates the applicability of the surveillance activities to Unit 1 and Unit 2 operations.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Radiation Monitoring Bases are discussed in Section 3/4.3.3.1 below.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip via P-16, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level and allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure or low compensated steamline pressure signals, reinstates steamline isolation on low compensated steamline pressure signals, and opens the accumulator discharge isolation valves. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure or low compensated steamline pressure signals, allows the manual block of steamline isolation on low compensated steamline pressure signals, and enables steam line isolation on high negative steam line pressure rate (when steamline pressure is manually blocked).

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips the turbine and the main feedwater pumps, and closes all feedwater isolation valves and feedwater control valves.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determine whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

PRESSURE TEMPERATURE LIMITS (Continued)

a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 621°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years (EFPY) of service life. The 32 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Tables B 3/4.4-1a and B 3/4.4-1b. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation

PRESSURE/TEMPERATURE LIMITS (Continued)

can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 32 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference

TABLE B 3/4.4-1a

REACTOR VESSEL TOUGHNESS (UNIT 1)

Component	Code	Grade	Cu (%)	P (%)	T_{NDT} (°F)	50 ft-lb 35 mil Temp. (°F)	RT_{NDT} (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
Closure head dome	R1616-1	A533B CL 1	0.07	0.018	-30	80	20	116	-
Closure head torus	R1615-1	A533B CL 1	0.04	0.010	-30	<30	-30	152	-
Closure head torus	R1615-2	A533B CL 1	0.11	0.012	-30	<30	-30	196	-
Closure head torus	R1615-3	A533B CL 1	0.07	0.011	-40	<20	-40	132	-
Closure head torus	R1615-4	A533B CL 1	0.13	0.018	-30	<50	-10	133	-
Closure head flange	R1602-1	A508 CL 2	0.05	0.007	0	<60	0	109	-
Vessel flange	R1601-1	A508 CL 2	0.02	0.017	-10	<50	-10	160.5	-
Inlet nozzle	R1613-1	A508 CL 2	-	0.009	-10	<50	-10	140	-
Inlet nozzle	R1613-2	A508 CL 2	-	0.013	0	<60	0	130.5	-
Inlet nozzle	R1613-3	A508 CL 2	0.09	0.009	-20	<40	-20	175	-
Inlet nozzle	R1613-4	A508 CL 2	-	0.006	20	<80	20	128	-
Outlet nozzle	R1614-1	A508 CL 2	-	0.006	10	<70	10	106	-
Outlet nozzle	R1614-2	A508 CL 2	-	0.006	0	<60	0	114	-
Outlet nozzle	R1614-3	A508 CL 2	-	0.009	-30	<30	-30	129	-
Outlet nozzle	R1614-4	A508 CL 2	-	0.006	10	<70	10	118	-
Nozzle shell	R1607-1	A533B CL 1	0.08	0.012	0	110	50	89	-
Nozzle shell	R1607-2	A533B CL 1	0.08	0.012	-20	110	50	85	-
Nozzle shell	R1607-3	A533B CL 1	0.07	0.010	-50	90	30	82	-
Inter. shell	R1606-1	A533B CL 1	0.04	0.009	-40	70	10	109.5	130
Inter. shell	R1606-2	A533B CL 1	0.04	0.008	-20	60	0	94	119
Inter. shell	R1606-3	A533B CL 1	0.05	0.007	-20	70	10	105.5	132
Lower shell	R1622-1	A533B CL 1	0.05	0.006	-30	30	-30	111	143
Lower shell	R1622-2	A533B CL 1	0.07	0.006	-30	30	-30	122	149
Lower shell	R1622-3	A533B CL 1	0.05	0.007	-30	30	-30	127	148
Bottom head torus	R1617-1	A533B CL 1	0.14	0.012	-50	<10	-50	143	-
Bottom head torus	R1618-1	A533B CL 1	0.08	0.015	-50	<10	-50	128	-
Inter. and lower shell vert. welds	G1.70	SAW	0.03	0.004	-50	<10	-50	*158	-
Inter. and lower shell girth weld	E3.13	SAW	0.03	0.007	-70	<10	-70	*100	-

*Normal to principal welding direction

TABLE B 3/4.4-1b

REACTOR VESSEL TOUGHNESS (UNIT 2)

Component	Code	Grade	Cu (%)	P (%)	T_{NDT} (°F)	50 ft-lb 35 mil Temp. (°F)	RT_{NDT} (°F)	Average Upper Shelf Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shelf Energy: Principal Working Direction (ft-lb)
Closure head dome	R3012-1	A533B CL 1	0.04	0.008	-40	<20	-40	144	-
Closure head torus	R3013-1	A533B CL 1	0.13	0.009	-30	40	-20	128	-
Closure head torus	R3013-2	A533B CL 1	0.13	0.009	-30	70	10	127	-
Closure head torus	R3013-3	A533B CL 1	0.15	0.012	-30	60	0	134	-
Closure head torus	R3013-4	A533B CL 1	0.15	0.012	-30	60	0	138	-
Closure head flange	R3002-1	A508 CL 2	0.06	0.008	-50	<10	-50	142	-
Vessel flange	R3001-1	A508 CL 2	0.04	0.008	-10	<50	-10	146	-
Inlet nozzle	R2011-1	A508 CL 2	0.10	0.011	-40	<20	-40	165	-
Inlet nozzle	R2011-2	A508 CL 2	0.10	0.011	-20	<40	-20	136	-
Inlet nozzle	R2011-3	A508 CL 2	0.12	0.009	-20	<40	-20	128	-
Inlet nozzle	R2011-4	A508 CL 2	0.11	0.009	-20	<40	-20	132	-
Outlet nozzle	R2012-1	A508 CL 2	-	0.006	10	<70	10	132	-
Outlet nozzle	R2012-2	A508 CL 2	-	0.007	10	<70	10	132	-
Outlet nozzle	R2012-3	A508 CL 2	-	0.004	0	<60	0	121	-
Outlet nozzle	R2012-4	A508 CL 2	-	0.007	10	<70	10	126	-
Nozzle shell	R2505-1	A533B CL 1	0.05	0.009	-40	60	0	114	-
Nozzle shell	R2505-2	A533B CL 1	0.07	0.008	-30	60	0	124	-
Nozzle shell	R2505-3	A533B CL 1	0.05	0.008	-50	50	-10	127	-
Inter. shell	R2507-1	A533B CL 1	0.04	0.006	-10	<50	-10	109	137
Inter. shell	R2507-2	A533B CL 1	0.05	0.006	-10	<50	-10	129	145
Inter. shell	R2507-3	A533B CL 1	0.05	0.005	-40	20	-40	122	149
Lower shell	R3022-1	A533B CL 1	0.03	0.002	-30	30	-30	124	141
Lower shell	R3022-2	A533B CL 1	0.04	0.003	-40	20	-40	118	141
Lower shell	R3022-3	A533B CL 1	0.04	0.004	-40	20	-40	123	126
Bottom Head Torus	R3020-1	A533B CL 1	0.11	0.009	-30	100	40	86	-
Bottom Head Torus	R3021-1	A533B CL 1	0.09	0.008	-60	0	-60	132	-
Inter. Shell Seams	G3.02	Sub Arc	0.05	0.004	-70	<10	-70	146*	-
Lower Shell and Inter. to Lower Girth Seam	E3.12	Sub Arc	0.05	0.008	-70	<10	-70	101*	-

*Normal to principal welding direction

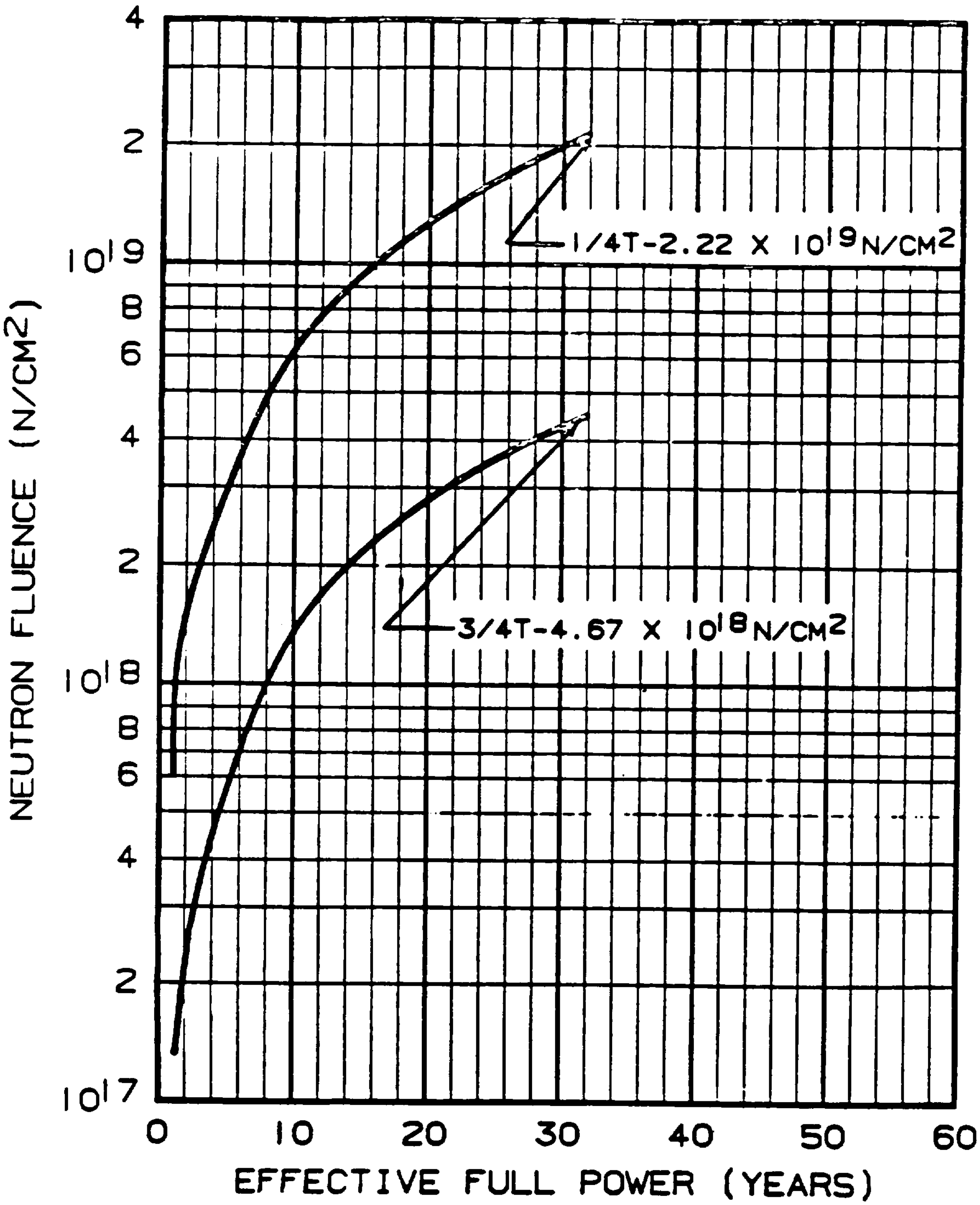


FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

PRESSURE/TEMPERATURE LIMITS (Continued)

fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the

PRESSURE/TEMPERATURE LIMITS (Continued)

1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and

PRESSURE/TEMPERATURE LIMITS (Continued)

increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.0 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the maximum credible mass injection flow rate due to the startup of a single HHSI pump plus 100 gpm net charging flow, while the RCS is in a water solid condition and the RCS temperature is between 350°F and 200°F.

For RCS temperatures less than 200°F, the maximum overpressure event consists of operating a centrifugal charging pump with complete termination of letdown and a failure of the charging flow control valve to the full flow condition.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all high head safety injection pumps while in MODE 5 and MODE 6 with the reactor vessel head on. All but one high head safety injection pump are required to be locked out in MODE 4. Technical Specifications also require lockout of the positive displacement pump and all but one charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The Maximum Allowed PORV Setpoint for the COMS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Winter 1975.

3/4.4.11 REACTOR VESSEL HEAD VENTS

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of at least two reactor vessel head vent paths ensures that the capability exists to perform this function.

The valve redundancy of the reactor vessel head 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 vessel head vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through three cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume represent a spread about an average value used in the safety analysis and have been demonstrated by sensitivity studies to vary the peak clad temperature by less than 20°F. The limit on accumulator pressure ensures that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened within one hour, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of three independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Each subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. One ECCS is assumed to discharge completely through the postulated break in the RCS loop. Thus, three trains are required to satisfy the single failure criterion. Note that the centrifugal charging pumps are not part of ECCS and that the RHR pumps are not used in the injection phase of the ECCS. Each ECCS subsystem and the RHR pumps and heat exchanges provide long-term core cooling capability in the recirculation mode during the accident recovery period.

When the RCS temperature is below 350°F, the ECCS requirements are balanced between the limitations imposed by the low temperature overpressure protection and the requirements necessary to mitigate the consequences of a LOCA below 350°F. At these temperatures, single failure considerations are not required because of the stable reactivity condition of the reactor and the limited core cooling requirements. Only a single Low Head Safety Injection pump is required to mitigate the effects of a large-break LOCA in this mode. However, two are

ECCS SUBSYSTEMS (Continued)

provided to accommodate the possibility that the break occurs in a loop containing one of the Low Head pumps. Low Head Safety Injection pumps are not required inoperable below 350°F because their shutoff head is too low to impact the low temperature overpressure protection limits.

Below 200°F (MODE 5) no ECCS pumps are required, so the High Head Safety Injection pumps are locked out to prevent cold overpressure.

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 flow testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA.

3/4.5.4 (This specification number is not used)

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, (2) the reactor will remain subcritical in the cold condition (68°F to 212°F) following a small break LOCA assuming complete mixing of the RWST, RCS, Spray Additive Tank, Containment Spray System and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), (3) the reactor will remain subcritical in cold condition following a large break LOCA (break flow area > 3.0 ft^2) assuming complete mixing of the RWST, RCS, Spray Additive Tank, Containment Spray System and ECCS water volumes and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO), and (4) long term subcriticality following a steamline break assuming ARI-1 and preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

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 RWST also ensure a pH value of between 7.5 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

This specification applies to the release of radioactive materials in liquid effluents from all units at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, <u>HASL-300</u> (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," <u>Anal. Chem. 40</u>, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report <u>ARH-SA-215</u> (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

DOSE (Continued)

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site.

3/4.11.1.3 LIQUID WASTE PROCESSING SYSTEM

The OPERABILITY of the Liquid Waste Processing System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Waste Processing System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks covered by this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Processing System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within

DOSE RATE (Continued)

the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1," July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure to man.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site.

3/4.11.2.4 GASEOUS WASTE PROCESSING SYSTEM

The OPERABILITY of the GASEOUS WASTE PROCESSING SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposure of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by the ODCM are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques" Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program given in the ODCM are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

BASES

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

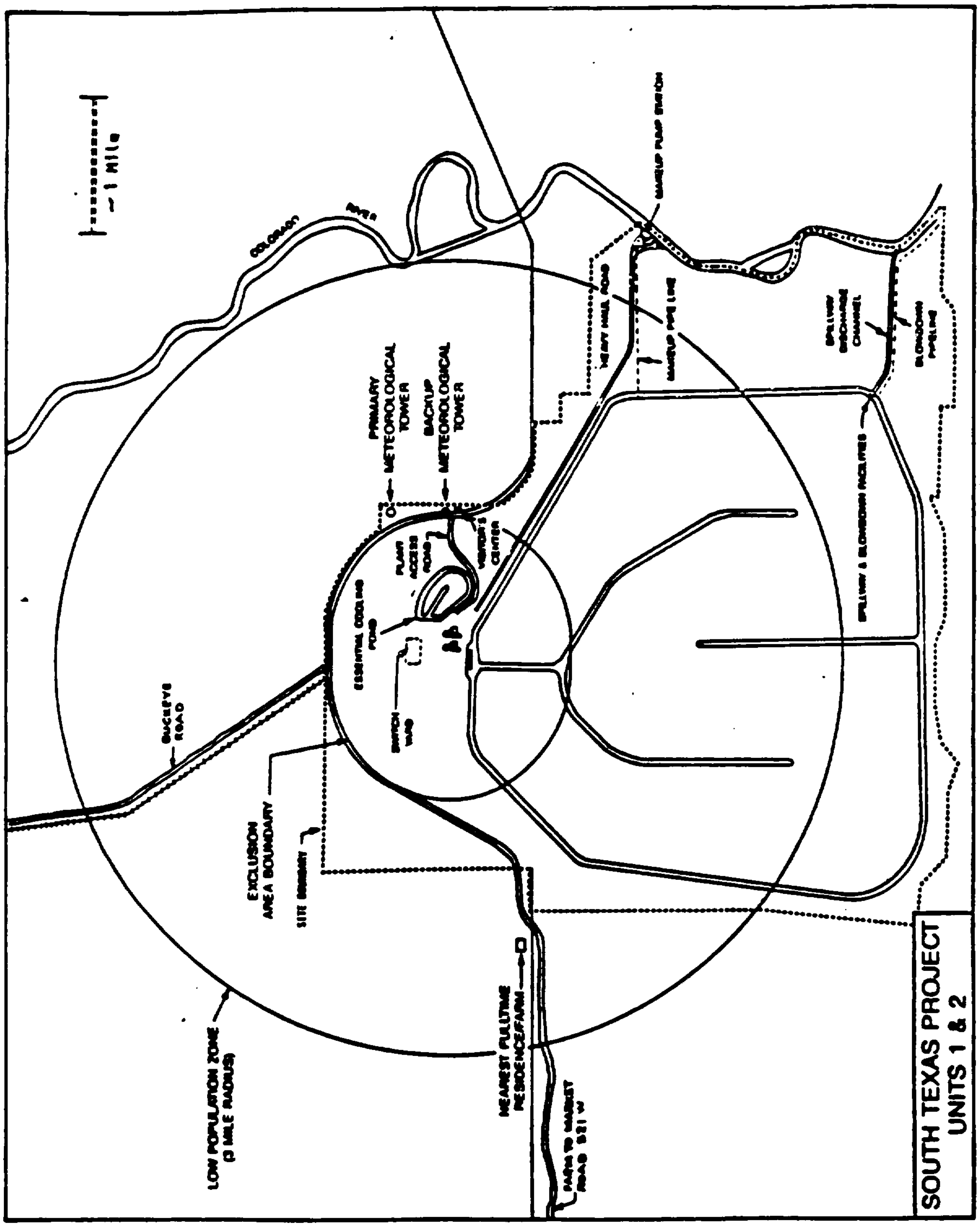


FIGURE 5.1-2
LOW POPULATION ZONE

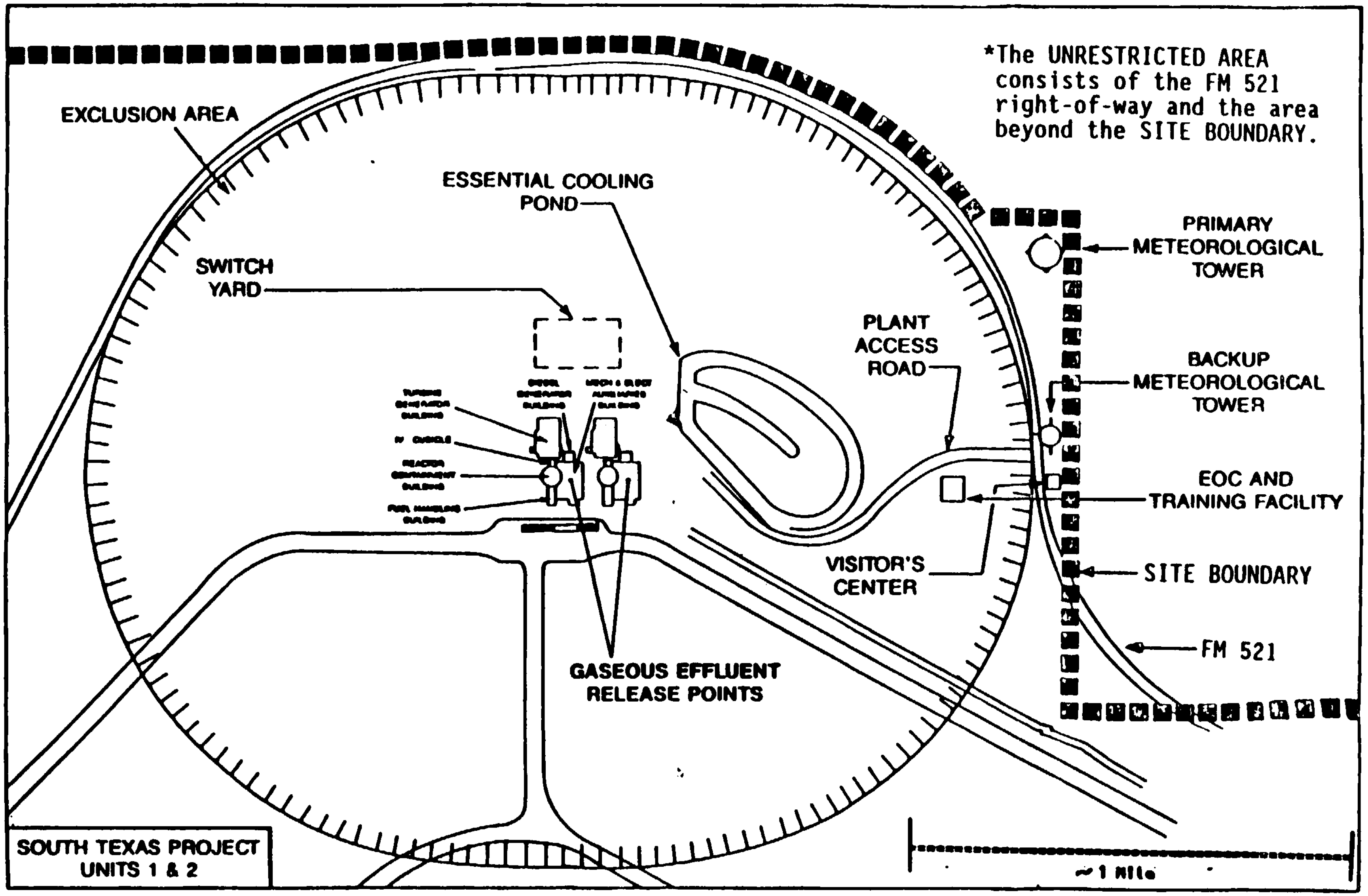
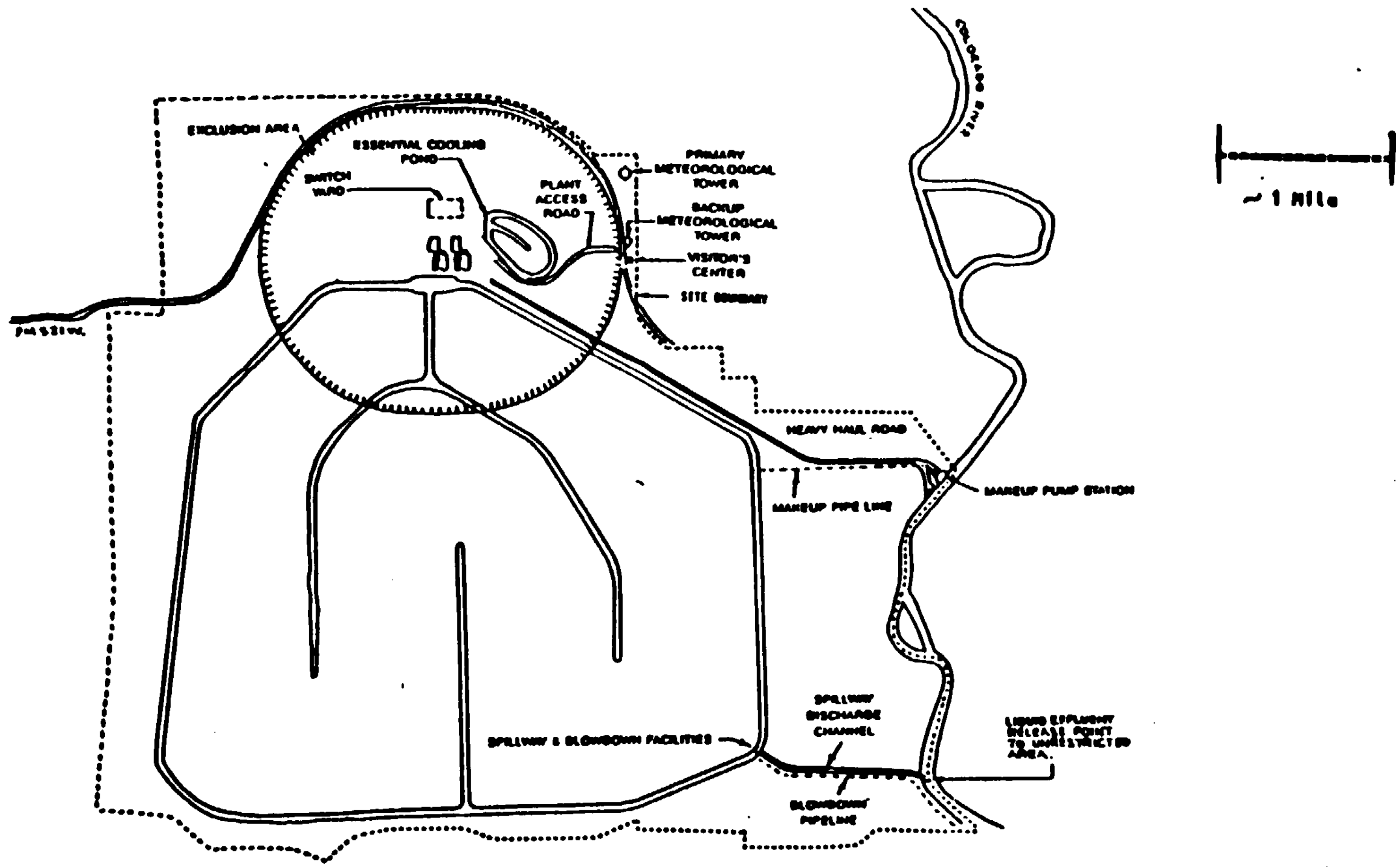


FIGURE 5.1-3

UNRESTRICTED AREA* AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS EFFLUENTS
(SEE FIGURE 5.1-4 FOR COMPLETE SITE BOUNDARY)



*The UNRESTRICTED AREA consists of the FM 521 right-of-way and the area beyond the SITE BOUNDARY.

SOUTH TEXAS PROJECT
UNITS 1 & 2

FIGURE 5.1-4

UNRESTRICTED AREA* AND SITE BOUNDARY FOR RADIOACTIVE LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 168 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 57 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 158.9 inches of absorber material. The absorber material shall be hafnium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,814 ± 100 cubic feet at a nominal T_{avg} of 561°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological towers shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of

0.0185 Δk for Region 1 uncertainties and tolerances and 0.0259 Δk for Region 2 uncertainties and tolerances.

b. A nominal 10.95 inches center to center distance between fuel assemblies in Region 1 of the storage racks and a nominal 9.15 inches center to center distance between fuel assemblies in Region 2 of the storage racks.

c. Neutron absorber (Boraflex) installed between spent fuel assemblies in the storage racks in Region 1 and Region 2.

d. Region 1 of the spent fuel storage racks can be used to store fuel which has a U-235 enrichment less than or equal to a nominal 4.5 weight percent. Region 2 can be used to store fuel which has achieved sufficient burnup such that storage in Region 1 is not required. The initial enrichment vs. burnup requirements of Figure 5.6.1 shall be met prior to storage of fuel assemblies in Region 2.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 62 feet-6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1969 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

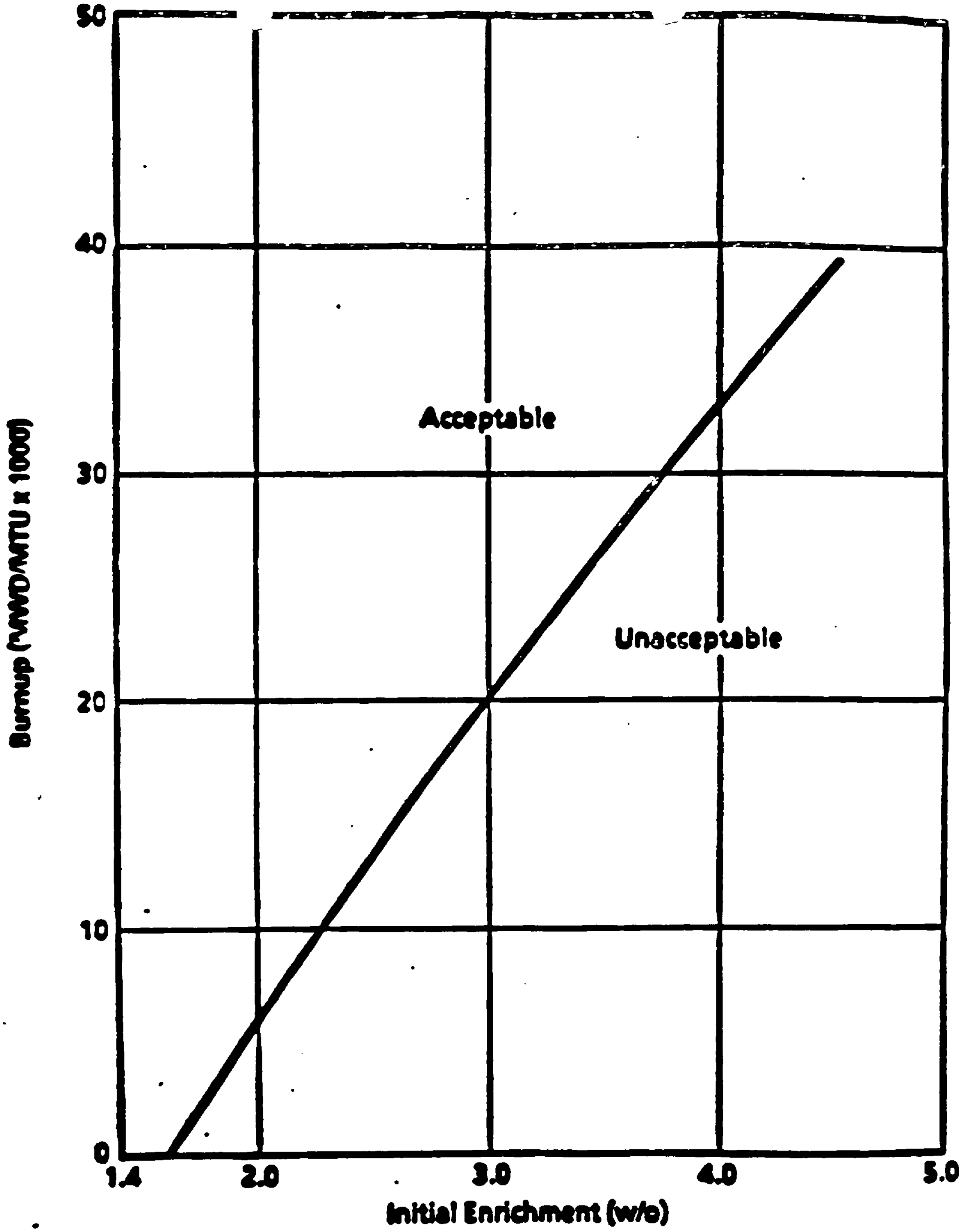


FIGURE 5.6-1

SOUTH TEXAS PROJECT SPENT FUEL RACKS
REGION 2 REQUIRED BURNUP AS A FUNCTION OF INITIAL ENRICHMENT

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 621°F.
	200 leak tests.	Pressurized to ≥ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to ≥ 3110 psig.
Secondary Coolant System	1 steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to ≥ 1600 psig.

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Group Vice President, Nuclear shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE AND ONSITE ORGANIZATIONS

6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or equivalent forms of documentation. These requirements shall be documented in the FSAR.

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President, Nuclear Plant Operations, shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF

6.2.2 The unit staff shall be as follows:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

UNIT STAFF (Continued)

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. A Health Physics Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site Fire Brigade of at least five members* shall be maintained on site at all times. The Fire Brigade shall not include the Shift Supervisor and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed (except for shift technical advisor personnel):

   1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

   2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

   3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

*The Health Physics Technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

g. Senior reactor operator licenses shall be held by:

Plant Operations Manager
Unit Operations Manager
Shift Supervisors
Unit Supervisors

Reactor operator licenses shall be held by:

Reactor Operators

TABLE 6.2-1
MINIMUM SHIFT CREW COMPOSITION
TWO UNITS WITH TWO SEPARATE CONTROL ROOMS

WITH THE OPPOSITE UNIT IN MODE 5 OR 6 OR DEFUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS	1*	1*
SRO	1	None
RO	2	1
RPO	2	2**
STA	1***	None

WITH THE OPPOSITE UNIT IN MODE 1, 2, 3, OR 4		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS	1*	1*
SRO	1	None
RO	2	1
RPO	2	1
STA	1* ***	None

SS - Shift Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
RPO - Reactor Plant Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

TABLE 6.2-1 (Continued)

TABLE NOTATIONS

*Individual may fill the same position on the opposite Unit.

**One of the two required individuals may fill the same position on the opposite Unit.

***The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Manager, Nuclear Safety Review Board.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Manager, Nuclear Safety Review Board.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 (Not Used)

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE(PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Member:	Plant Superintendent
Member:	Technical Services Manager
Member:	Plant Operations Manager
Member:	Plant Engineering Manager
Member:	Maintenance Manager
Member:	Quality Engineering Manager

The PORC Chairman shall be appointed in writing from among these members by the Plant Manager, except for the Quality Engineering Manager. If the Technical Services Manager does not meet the qualifications of a Radiation Protection Manager as defined in Regulatory Guide 1.8 (Personnel Selection and Training-Revision 1-R), then the PORC composition will include the Health Physics Manager.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the Plant Manager to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and three other members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of all safety-related station administrative procedures and changes thereto.

b. Review of safety evaluations for (1) procedures, (2) changes to procedures, structures, components, or systems, and (3) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question.

c. Review of proposed (1) procedures, (2) changes to procedures, structures, components, or systems, and (3) tests or experiments which may involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Review of all programs required by Specification 6.8 and changes thereto.

e. Review of proposed changes to the Technical Specifications or the Operating License.

f. Review of all REPORTABLE EVENTS.

g. Review of reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment or systems that affect nuclear safety.

h. Review of reports of unanticipated deficiencies in the design or operation of structures, systems, or components that affect nuclear safety.

i. Review of the Security Plan and implementing procedures and changes thereto.

j. Review of the Emergency Plan and implementing procedures and changes thereto.

k. Review of the PROCESS CONTROL PROGRAM and implementing procedures and changes thereto.

l. Review of the OFFSITE DOSE CALCULATION MANUAL and implementing procedures and changes thereto.

m. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager or the Nuclear Safety Review Board (NSRB).

n. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Plant Manager and to the NSRB.

o. Reports of violations of codes, regulations, orders, Technical Specifications, or Operating License requirements having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.

p. Review of the Fire Protection Program, quality-related implementing procedures and changes thereto.

RESPONSIBILITIES (Continued)

6.5.1.7 The PORC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6a. through e. prior to their implementation, and items considered under Specification 6.5.1.6i. through l.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. through e. and o. constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Group Vice President-Nuclear and the Nuclear Safety Review Board of disagreement between the PORC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Group Vice President-Nuclear and the Nuclear Safety Review Board.

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB)

FUNCTION

6.5.2.1 The NSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering,
h. Civil engineering,
i. Training,
j. Nuclear assurance,
k. Nuclear licensing,
l. Plant security, and
m. Environmental impact.

The NSRB shall report to and advise the Group Vice President-Nuclear on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

COMPOSITION

6.5.2.2 The NSRB shall be composed of the following, and other members shall be appointed in writing by the Group Vice President, Nuclear

Chairman	General Manager, NSRB
Member:	General Manager, South Texas Project Management
Member:	Vice President, Nuclear Plant Operations
Member:	General Manager, Nuclear Assurance
Member:	General Manager, South Texas Project Operations Support

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the Group Vice President-Nuclear to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NSRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSRB Chairman to provide expert advice to the NSRB.

MEETING FREQUENCY

6.5.2.5 The NSRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the NSRB necessary for the performance of the NSRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least a majority of NSRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The NSRB shall be responsible for the review of:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

REVIEW (Continued)

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PORC.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training, and qualifications of the entire unit staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

AUDITS (Continued)

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months; and

k. Any other area of unit operation considered appropriate by the NSRB or the Group Vice President-Nuclear.

RECORDS

6.5.2.9 Records of NSRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRB meeting shall be prepared, approved, and forwarded to the Group Vice President-Nuclear within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Group Vice President-Nuclear within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Group Vice President-Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities that affect nuclear safety shall be conducted as follows:

a. Procedures required by Specification 6.8, and other procedures that affect nuclear safety, and changes thereto, shall be prepared, reviewed, and approved. Each such procedure, or change thereto, shall be reviewed by an individual/group other than the individual/group who prepared the procedure, or change thereto, but who may be from the same organization as the individual/group who prepared the procedure, or change thereto. Procedures other than station administrative procedures shall be approved by the Plant Manager, Plant Superintendent, or the head of the responsible department prior to implementation. The Plant Manager shall approve station administrative procedures, security plan implementing procedures, and emergency plan implementing procedures. Temporary changes to procedures, which clearly do not change the intent of the approved procedures, shall be approved prior to implementation by two members of the plant staff, ACTIVITIES (Continued)

at least one of whom holds a Senior Reactor Operator's License. Changes to procedures that may involve a change to the intent of the original procedure shall be approved by the individual authorized to approve the procedure prior to implementation of the change.

b. Proposed changes or modifications to safety-related structures, systems, and components shall be reviewed as designated by the Plant Manager. Each such modification shall be reviewed by an individual/group other than the individual/group who designed the modification, but who may be from the same organization as the individual/group who designed the modification. Proposed modifications to safety-related structures, systems, and components shall be approved by the Plant Manager prior to implementation.

c. Proposed tests and experiments that affect nuclear safety and that are not addressed in the Final Safety Analysis Report shall be prepared, reviewed, and approved prior to implementation. Each such test or experiment shall be reviewed by an individual/group other than the individual/group who prepared the test or experiment but who may be from the same organization as the individual/group who prepared the test or experiment. Proposed tests and experiments shall be approved by the Plant Manager.

d. Individuals responsible for reviews performed in accordance with Specification 6.5.3.1 (a) through (c) shall be members of the plant management staff previously designated by the Plant Manager. Each review shall include a determination of whether or not additional, cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by qualified personnel of the appropriate discipline.

e. Each review will include a determination of whether or not an unreviewed safety question is involved. Pursuant to 10 CFR 50.59, NRC approval of items involving an unreviewed safety question will be obtained prior to Plant Manager approval for implementation.

6.5.3.2 Records of the above activities shall be provided to the Plant Manager, PORC, and/or the NSRB as necessary for required reviews.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PORC, and the results of this review shall be submitted to the NRSB and the Group Vice President-Nuclear.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

SAFETY LIMIT VIOLATION (Continued)

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Group Vice President-Nuclear and the NSRB shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence;

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB, and the Group Vice President-Nuclear within 14 days of the violation; and

d. Operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

g. Quality Assurance Program for effluent and environmental monitoring; and

h. Fire Protection Program implementation.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed and approved prior to implementation and reviewed periodically as set forth in Specification 6.5.3 and administrative procedures.

6.8.3 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the containment spray, Safety Injection, containment hydrogen monitoring, post-accident sampling and primary sampling. The program shall include the following:

HIGH RADIATION AREA (Continued)

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   3) Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Changes to Part A shall be submitted to and approved by the NRC staff prior to implementation.

b. Changes to Part B shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

OFFSITE DOSE CALCULATION MANUAL (ODCM) (Continued)

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or Setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

c. Changes to Part B shall become effective upon review and acceptance by the PORC.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 4 TO

FACILITY OPERATING LICENSE NO. NPF-76

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

1.0 INTRODUCTION

By application dated November 7, 1988, Houston Lighting & Power Company, et al., (the licensee) requested changes to the Technical Specifications (TS) (Appendix A to Facility Operating License No. NPF-76) for South Texas Project, Unit 1. The proposed modification would change the Unit 1 TS to the Combined TS for Units 1 and 2, add the requirement that the positive displacement pump be in a lock-out condition before reaching Cold Overpressurization Mitigation System activation conditions, add a reactor coolant pump isolation header pressure interlock, and make modifications to the administrative section of the TS.

2.0 DISCUSSION AND EVALUATION

The staff has reviewed each of the areas of the proposed amendment and the licensee's no significant hazards consideration determination. The review of each of these areas is discussed below.

The South Texas Project Unit 1 operating license includes the TS for the operation of Unit 1. At the time Unit 2 receives an operating license, Houston Lighting & Power Company (HL&P) will receive TS that are applicable for both units, i.e., Combined TS. To implement the Combined TS on Unit 1, the Unit 1 license requires an amendment. The proposed changes are administrative in nature only; the units are identical. No hardware or operational changes are being made as a result of the amendment.

The section of the proposed amendment regarding the positive displacement pump (PDP) adds to the TS the placement of the PDP in a lock-out condition in Modes 4, 5 and 6 before reaching a cold overpressure mitigation system activation condition. The change takes into account the fact that the PDP will be operated infrequently for hydrostatic testing purposes. Since the PDP flow rate is small and it is operated infrequently, the probability of an overpressurization is slightly increased. This increase and the fact that it is not appropriate that the pump be locked out during hydrostatic testing justifies the inclusion of an exception in the TS. This exception occurs when the reactor vessel head is removed, when the PDP is required for hydrostatic testing, or when both centrifugal charging pumps are inoperable.

The section of the proposed amendment regarding the reactor coolant pump (RCP) seal isolation proposes to incorporate the RCP seal isolation charging header pressure interlock into the TS. Because of the design at STP (separate charging and safety injection pumps) it is preferable to maintain seal flow to the RCP seals after a containment isolation. To accomplish this, the STP design incorporates an interlock which allows the seal injection isolation valves to remain open as long as charging header pressure is maintained.

The RCP Seal Injection Isolation function monitors the charging header pressure and provides an isolation signal to the RCP seal injection containment isolation valves if a low charging header pressure occurs with a Phase "A" containment isolation. Header pressure is detected by a single instrument channel. The proposed changes to the TS address the operability and surveillance requirements for this channel and the containment isolation valves.

Four changes were proposed regarding the administrative section of the TS. The first change involves the composition of the Plant Operations Review Committee (PORC). The change requires that if the Technical Services Manager does not meet the qualifications of a Radiation Protection Manager, the PORC will be augmented by a member who does meet the qualifications. The second change further defines the quorum requirements for the Nuclear Safety Review Board by indicating that a majority of the board members must be present for a quorum to exist. The third change specifies the minimum approval authority for plant procedures. For procedures other than station administrative procedures, the Plant Manager, Plant Superintendent or other responsible department head will approve procedures prior to implementation. The fourth change specifies that procedures will be reviewed periodically. The changes are administrative in nature only and involve no hardware or operation changes.

None of the modifications listed above involve significant increases in the probability or consequences of an accident, create the possibility of a new or different kind of accident or involve a significant decrease in the margin of safety.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment relates to changes in installation or use of a facility component located within the restricted area and to changes in recordkeeping, or administrative procedures or requirements. The staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

Based upon its evaluation of the proposed changes to the South Texas Project, Unit 1, Technical Specifications, the staff has concluded that: there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The staff, therefore, concludes that the proposed changes are acceptable, and are hereby incorporated into the South Texas Project, Unit 1 Technical Specifications.

Date: December 29, 1988

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