SCE&G
A SCANA COMPANY

January 8, 2001
RC-01-0007

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Ms. K. R. Cotton

Stephen A. Byrne
Vice President
Nuclear Operations
803.345.4622

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
TECHNICAL SPECIFICATION CHANGE REQUEST
TSP 99-0263 Administrative Change Request

South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS). This request is being submitted pursuant to 10 CFR 50.90.

South Carolina Electric & Gas Co
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, South Carolina
29065

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The Reactor Trip System Instrumentation trip setpoints and associated Bases are being revised to remove conflicting values and correct the information provided in the Bases.

Several editorial changes are included in this request to 3/4.8.1.1 that will provide some flexibility in testing Emergency Diesel Generator fuel oil per ASTM standards. These changes will permit the testing to be based upon the standards as opposed to being in accordance with the standards.

Section 6.5, Plant Safety Review Committee, is being revised to remove the requirement for review of Fire Protection Program changes, based on guidance in Regulatory Information Summary (RIS) 99-02.

Section 6.12, High Radiation Areas, is being revised to change the definitions for high radiation area and very high radiation area to be consistent with 10 CFR 20.1003. This change affects the footnotes on pages 6-18 and 6-19.

A BASES change is included that will revise BASES page B 2-8 to correct the description for Reactor Trip System Interlock P-7 by removing the reactor trip on turbine trip for P-7. The reactor trip on turbine trip interlock is actually provided by interlock P-9 (50% power) and was inadvertently omitted during the review for Amendment 34.

Other BASES changes for page B 2-8 involve Interlock P-6 and P-10. Amendment 65 approved the use of the Gamma-Metrics System for Nuclear Instrumentation; the Reactor Trip System Interlocks were overlooked during the review process. There is no longer any need for an interlock to automatically de-energize the detectors before or after they are over-ranged.

Additionally, the BASES section for the ECCS Accumulators (B3.5.1) is being revised to clarify the operational constraints on the accumulator isolation valves and the applicability of portions of IEEE 279-1971 to these valves. Currently, the BASES describe these valves as "operating bypasses in the context of IEEE 279-1979". This function is not required for these valves due to operational philosophy and procedures, therefore the BASES needs to clarify this fact. This change is the result of Westinghouse vendor bulletin (NSAL-97-003).

Multiple editorial changes are also proposed to correct misspelled words, incorrect case, obviously missing words, and section titles. These changes were determined to be within the scope of an editorial change.

The TS change request is contained in the following attachments:

Attachment I	Explanation of Changes Summary Marked-up Technical Specification Pages Revised Technical Specification Pages
Attachment II	Safety Evaluation
Attachment III	No Significant Hazards Evaluation

This proposed amendment has been reviewed and approved by the Plant Safety Review Committee and the Nuclear Safety Review Committee.

This submittal is requested to be approved by at your earliest convenience with a 30-day implementation period.

There are no other TS changes in process that will affect or be affected by this change request.

There are no significant changes to any FSAR or FPER sections.

There are no commitments proposed as a result of this change request.

I certify under penalty of perjury that the foregoing is true and correct.

Should you have questions, please call Mr. Philip A. Rose at (803) 345-4052.

Very truly yours,

Stephen A. Byrne

PAR/SAB/dr
Attachments (3)

c: N. O. Lorick
N. S. Carns
T. G. Eppink (w/o Attachment)
R. J. White
L. A. Reyes
K. R. Cotton
NRC Resident Inspector
Paulett Ledbetter
J. B. Knotts, Jr.
T. P. O'Kelley
RTS (TSP 99-0263)
File (813.20)
DMS (RC-01-0007)

STATE OF SOUTH CAROLINA :
: TO WIT :
COUNTY OF FAIRFIELD :

I hereby certify that on the 8th day of January 2001, before me, the subscriber, a Notary Public of the State of South Carolina personally appeared Stephen A. Byrne, being duly sworn, and states that he is Vice President, Nuclear Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal

Notary Public

My Commission Expires

July 13 2005
Date

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revision is indicated by a marginal line.

Remove Page	Insert Page
2-1	2-1
2-7	2-7
3/4 1-17	3/4 1-17
3/4 3-8	3/4 3-8
3/4 3-12	3/4 3-12
3/4 4-9	3/4 4-9
3/4 6-17	3/4 6-17
3/4 6-22	3/4 6-22
3/4 7-18	3/4 7-18
3/4 7-21	3/4 7-21
3/4 8-3	3/4 8-3
3/4 8-4	3/4 8-4
3/4 8-6	3/4 8-6
3/4 9-8	3/4 9-8
3/4 11-5	3/4 11-5
B 2-8	B 2-8
B 3/4 3-2	B 3/4 3-2
B 3/4 5-1	B 3/4 5-1
B 3/4 7-3	B 3/4 7-3
B 3/4 9-2	B 3/4 9-2
6-5	6-5
6-9	6-9
6-12a	6-12a
6-18	6-18
6-19	6-19

SCE&G -- EXPLANATION OF CHANGES

Page	Affected Section	Bar #	Description of Change	Reason for Change
2-1	2.1.1	1	Change word Figures with word Figure	Amendment 75 deleted reference to other figures
2-7	Table 2.2-1	1	Delete trip Setpoints and Allowable Values for P-7 Interlock	Remove conflicting values in P-7 as identified in CER 99-1191 and follow guidance of NUREG 1431, Revision 1
3/4 1-17	3.1.3.2.a.2	1	Change "TO" to lower case	Only Defined words should be upper case
3/4 3-8	Table 3.3-1	1	Capitalize "Minimum"	Provide consistency with the rest of the notes on this page
3/4 3-12	Table 4.3-1	1	Delete reference to note (8)	Note 8 was deleted by Amendment 101
3/4 4-9	3.4.3	1	Change kw to KW	Assure consistency in nomenclature
3/4 6-17	4.6.4.2	1	Change "AT LEAST ONCE PER 18 MONTHS BY" to lower case	Only defined words should be upper case
3/4 6-22	4.6.5.2.a	1	Change Kw to KW	Assure consistency in nomenclature
3/4 7-18	4.7.7.f.3	1	Pluralize "direction"	Verifying motion in more than 1 direction
3/4 7-21	Table 4.7-2	1	Change word ration to ratio	Correct typographical error

Page	**Affected Section**	**Bar #**	**Description of Change**	**Reason for Change**
3/4 8-3	4.8.1.1.2.a.4	1	Change kW to KW	Assure consistency in nomenclature
	4.8.1.2.d	2	Remove the words "in accordance with" and replace with "based on"	Provide flexibility in the applicability of portions of specific ASTM Standards used in Surveillance Tests
	4.8.1.1.2.d.1	3	Remove the words "in accordance with" and replace with "based on"	Provide flexibility in the applicability of portions of specific ASTM Standards used in Surveillance Tests
	4.8.1.1.2.d.1.a 4.8.1.1.2.d.1.b	4 5	Changed a. and b. to a) and b)	Correct punctuation error
3/4 8-4	4.8.1.1.2.d.1.c 4.8.1.1.2.d.1.d	1 2	Changed c. and d. to c) and d)	Correct punctuation error
	4.8.1.1.2.d.1.d	2 cont.	Remove the words "in accordance with" and replace with "based on"	Provide flexibility in the applicability of portions of specific ASTM Standards used in Surveillance Tests
	4.8.1.1.2.d.2	3	Remove the words "in accordance with" and replace with "based on"	Provide flexibility in the applicability of portions of specific ASTM Standards used in Surveillance Tests
	4.8.1.1.2.e	4	Remove the words "in accordance with" and replace with "based on"	Provide flexibility in the applicability of portions of specific ASTM Standards used in Surveillance Tests
	4.8.1.1.2.f.2	5	Change kW to KW	Assure consistency in nomenclature

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 8-4 cont.	4.8.1.1.2.g.2	6	Add units to load rejection of 729	Amendment 77 changed value and inadvertently left off the units
	4.8.1.1.2.g.3	7	Change kW to KW	Assure consistency in nomenclature
3/4 8-6	4.8.1.1.2.g.7.a	1	Change kw to KW	Assure consistency in nomenclature
	4.8.1.1.2.g.7.b	2	Change kw to KW	Assure consistency in nomenclature
	4.8.1.1.2.g.8	3	Change kw to KW	Assure consistency in nomenclature
	4.8.1.1.2.g.13 a and b	4	Changed a. and b. to a) and b)	Correct punctuation error
3/4 9-8	3.9.7.2.a	1	Add the word "vessel"	Word inadvertently omitted on initial issue of TS
3/4 11-5	4.11.2.6	1	Replace word one with once	Word inadvertently changed by Amendment 133
B 2-8	B 2.2.1 P-6	1	Removed discussion of de-energizing power supply to the detectors	Amendment 65 approved Gamma-Metrics system but omitted the Reactor Trip System Interlocks change

Page	**Affected Section**	**Bar #**	**Description of Change**	**Reason for Change**
B 2-8 cont.	B 2.2.1 P-7	2	a. Removed discussion of reactor trip on turbine trip for P-7 b. Add discussion to describe why Trip Setpoints and Allowable Values are not applicable to Interlock P-7 in Table 2.2-1	a. Amendment 34 incorporated P-9, which gives reactor trip on turbine trip above 50% RTP. b. Conflicting values in Table for P-7 as identified in CER 99-1191 and follow guidance in NUREG 1431, Revision 1
	B 2.2.1 P-10	3	Removed discussion of de-energizing power supply to the Source Range high voltage detectors	Amendment 65 approved Gamma-Metrics system but omitted the Reactor Trip System Interlocks change
B 3/4 3-2	B 3/4.3.1, B 3/4.3.2	1	Replace word "protection" with "trip"; remove word "system"	Assure consistency in BASES titles
B 3/4.5-1	B 3/4.5.1	1	Adding discussion of accumulator isolation valves "operating bypasses"	Westinghouse NSAL-97-003 technical bulletin - verbatim compliance issue
B 3/4 7-3	B 3/4.7.2	1	Change word "valves" to values	Correct typographical error from initial issue of TS
B 3/4 9-2	B 3/4.9.7	1	Correct spelling of word thru	Correct typographical error from initial issue of TS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

TABLE2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
18.	Safety Injection Input from ESF	NA	NA
19.	Reactor Trip System Interlocks		
	A. Intermediate Range Neutron Flux, P-6	$\geq 7.5 \times 10^{-6}$% indication	$\geq 4.5 \times 10^{-6}$% indication
	B. Low Power Reactor Trips Block, P-7 a. P-10 input	~~≤10% of RTP~~ N/A	~~≤12.2% of RTP~~ N/A
	b. P-13 input	~~≤10% turbine impulse pressure equivalent~~ N/A	~~≤12.2% of turbine impulse pressure equivalent~~ N/A
	C. Power Range Neutron Flux P-8	≤38% of RTP	≤40.2% of RTP
	D. Low Setpoint Power Range Neutron Flux, P-10	≥10% of RTP	≥7.8% of RTP
	E. Turbine Impulse Chamber Pressure, P-13	≤10% turbine impulse pressure equivalent	≤12.2% turbine pressure equivalent
	F. Power Range Neutron Flux, P-9	≤50% of RTP	≤52.2% of RTP
20.	Reactor Trip Breakers	NA	NA
21.	Automatic Actuation Logic	NA	NA

RTP - RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
18.	Safety Injection Input from ESF	NA	NA
19.	Reactor Trip System Interlocks		
	A. Intermediate Range Neutron Flux, P-6	$\geq 7.5 \times 10^{-6}$ % indication	$\geq 45 \times 10^{-6}$ % indication
	B. Low Power Reactor Trips Block, P-7		
	a. P-10 input	N/A	N/A
	b. P-13 input	N/A	N/A
	C. Power Range Neutron Flux P-8	≤ 38% of RTP	≤ 40.2% of RTP
	D. Low Setpoint Power Range Neutron Flux, P-10	≥ 10% of RTP	≥ 7.8% of RTP
	E. Turbine Impulse Chamber Pressure, P-13	≤ 10% turbine impulse pressure equivalent	≤ 12.2% turbine pressure equivalent
	F. Power Range Neutron Flux, P-9	≤ 50% of RTP	≤ 52.2% of RTP
20.	Reactor Trip Breakers	NA	NA
21.	Automatic Actuation Logic	NA	NA

RTP - RATED THERMAL POWER

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown and control rod position indication system and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours.

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown and control rod position indication system and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

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

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

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SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - With the number of OPERABLE Channels less than the Total Number of Channels operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker OPERABLE status.

ACTION 12 With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - With the number of OPERABLE Channels less than the Total Number of Channels operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker OPERABLE status.

ACTION 12 With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1288 cubic feet, (92% of indicated span) and at least two groups of pressurizer heaters each having a capacity of at least 125 ~~kw.~~ KW

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days. 14

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1288 cubic feet, (92% of indicated span) and at least two groups of pressurizer heaters each having a capacity of at least 125 KW.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE.*

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 either:

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

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

c. Isolate 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.

The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment isolation valve 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.

4.6.4.2 Each containment 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 containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Reactor Building Purge and Exhaust isolation test signal, each Purge and Exhaust valve actuates to its isolation position.

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

CONTAINMENT SYSTEMS

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE.*

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 either:

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

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

c. Isolate 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.

The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment isolation valve 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.

4.6.4.2 Each containment 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 containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Reactor Building Purge and Exhaust isolation test signal, each Purge and Exhaust valve actuates to its isolation position.

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control .

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 Two independent post accident hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying, during a recombiner system functional test, that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 KW.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

   2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

   3. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 Two independent post accident hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying, during a recombiner system functional test, that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 KW.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

   2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

   3. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests (Continued)

When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers of each type. The representative sample shall be weighted to include more snubbers from severe service areas such as near heavy equipment. Snubbers placed in the same location as snubbers which failed the previous functional test shall be included in the next test lot if the failure analysis shows that failure was due to location.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers, may be tested to verify only that activation takes place in both directions of travel.

2. Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range.

3. Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

5. Fasteners for attachment of the snubber to the component and to the snubber anchorage are secure.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

e. Functional Tests (Continued)

When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers of each type. The representative sample shall be weighted to include more snubbers from severe service areas such as near heavy equipment. Snubbers placed in the same location as snubbers which failed the previous functional test shall be included in the next test lot if the failure analysis shows that failure was due to location.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

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2. Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range.

3. Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

5. Fasteners for attachment of the snubber to the component and to the snubber anchorage are secure.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

TABLE 4.7-2
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

TABLE NOTATION

(1) The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that internal. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

(2) Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that includes a fractional value of unacceptable snubbers as determined by interpolation.

(3) If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

(4) If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

(5) If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ration of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

(6) The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

TABLE 4.7-2
SNUBBER VISUAL INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	26
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

TABLE NOTATION

(1) The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that internal. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

(2) Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that includes a fractional value of unacceptable snubbers as determined by interpolation.

(3) If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

(4) If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

(5) If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

(6) The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each EDG shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank and fuel storage tank.

2. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

3. Verifying the diesel generator can start* and accelerate to synchronous speed (504 rpm) with generator voltage and frequency at 7200 ± 720 volts and 60 ± 1.2 Hz.

4. Verifying the generator is synchronized, gradually loaded* to an indicated 4150-4250 ~~kW~~** KW and operates for at least 60 minutes.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by removing accumulated water from the day tank.

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

d. By sampling new fuel oil ~~in accordance with~~ based on the applicable ASTM standard prior to addition to storage tanks and:

1. By verifying ~~in accordance with~~ based on the tests specified in the applicable ASTM standard prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

b) A kinematic viscosity of 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but not less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each EDG shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank and fuel storage tank.

2. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

3. Verifying the diesel generator can start* and accelerate to synchronous speed (504 rpm) with generator voltage and frequency at 7200 ± 720 volts and 60 ± 1.2 Hz.

4. Verifying the generator is synchronized, gradually loaded* to an indicated 4150-4250 KW** and operates for at least 60 minutes.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by removing accumulated water from the day tank.

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

d. By sampling new fuel oil based on the applicable ASTM standard prior to addition to storage tanks and:

1. By verifying based on the tests specified in the applicable ASTM standard prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

b) A kinematic viscosity of 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but not less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance with proper color when tested ~~in accordance with~~ based on the applicable ASTM standard.

2. By verifying within 30 days of obtaining the sample that the specified properties are met when tested ~~in accordance with~~ based on the applicable ASTM standard.

e. At least once every 31 days by obtaining a sample of fuel oil ~~in accordance with~~ based on the applicable ASTM standard, and verifying that total contamination is less than 10 mg/liter when checked ~~in accordance with~~ based on the applicable ASTM standard.

f. At least once per 184 days by:

1. Starting and accelerating the EDG to synchronous speed (504 rpm) with generator voltage and frequency at 7200 ± 720 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The EDG shall be started for this test by using one of the following signals:

a) Simulated loss of offsite power by itself.
b) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
c) An ESF actuation test signal by itself.
d) Simulated degraded offsite power by itself.
e) Manual.

2. The generator shall be manually synchronized, loaded to an indicated 4150-4250 ~~kW~~ KW** in less than or equal to 60 seconds, and operate for at least 60 minutes.

g. At least once every 18 months by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying that on rejection of a load of greater than or equal to 729 KW, the voltage and frequency are maintained at 7200 ± 720 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the generator capability to reject a load of 4250 ~~kW~~ KW without tripping. The generator voltage shall not exceed 7920 volts during and following the load rejection.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance with proper color when tested based on the applicable ASTM standard.

2. By verifying within 30 days of obtaining the sample that the specified properties are met when tested based on the applicable ASTM standard.

e. At least once every 31 days by obtaining a sample of fuel oil based on the applicable ASTM standard, and verifying that total contamination is less than 10 mg/liter when checked based on the applicable ASTM standard.

f. At least once per 184 days by:

1. Starting and accelerating the EDG to synchronous speed (504 rpm) with generator voltage and frequency at 7200 ± 720 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The EDG shall be started for this test by using one of the following signals:

a) Simulated loss of offsite power by itself.
b) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
c) An ESF actuation test signal by itself.
d) Simulated degraded offsite power by itself.
e) Manual.

2. The generator shall be manually synchronized, loaded to an indicated 4150-4250 KW** in less than or equal to 60 seconds, and operate for at least 60 minutes.

g. At least once every 18 months by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying that on rejection of a load of greater than or equal to 729 KW, the voltage and frequency are maintained at 7200 ± 720 volts and frequency at 60 ± 1.2Hz.

3. Verifying the generator capability to reject a load of 4250 KW without tripping. The generator voltage shall not exceed 7920 volts during and following the load rejection.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

7. Verifying the EDG operates for at least 24 hours.

   a) The EDG shall be loaded to the continuous rating (4150-4250 KW**) for the time required to reach engine temperature equilibrium, at which time the EDG shall be loaded to an indicated target value of 4676 KW (between 4600-4700 KW**) and maintained for 2 hours.

   b) During the remaining 22 hours of this test, the EDG shall be loaded to an indicated 4150-4250 KW**.

   c) During this test the steady state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

8. Verifying that the auto-connected loads to each EDG do not exceed the 2000 hour rating of 4548 KW.

9. Verifying the EDG's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10. Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

12. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) Barring Device

    b) Remote-Local-Maintenance Switch

14. Verifying that within 5 minutes of operating the diesel generator for at least 1 hour at a load of 4150-4250 KW** the diesel starts on the auto-start signal (Loss of Off-Site Power signal), energizes the emergency busses with permanently connected loads

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

7. Verifying the EDG operates for at least 24 hours

   a) The EDG shall be loaded to the continuous rating (4150-4250 KW**) for the time required to reach engine temperature equilibrium, at which time the EDG shall be loaded to an indicated target value of 4676 KW (between 4600-4700 KW **) and maintained for 2 hours.

   b) During the remaining 22 hours of this test, the EDG shall be loaded to an indicated 4150-4250 KW **.

   c) During this test the steady state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

8. Verifying that the auto-connected loads to each EDG do not exceed the 2000 hour rating of 4548 KW.

9. Verifying the EDG's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10. Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

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14. Verifying that within 5 minutes of operating the diesel generator for at least 1 hour at a load of 4150-4250 KW** the diesel starts on the auto-start signal (Loss of Off-Site Power signal), energizes the emergency busses with permanently connected loads

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

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

ACTION:

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

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.2 At least one residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

*Prior to initial criticality the residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

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

ACTION:

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

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.2 At least one residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

* Prior to initial criticality the residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 131,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. **With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.**

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least one once per 24 hours when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 131,000 curies noble gases (considered as Xe-133) .

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range reactor trip. ~~and de-energizing of the high voltage to the detectors.~~ On decreasing power, Source Range level trips are automatically reactivated. ~~and high voltage restored.~~

P-7 On increasing power P-7 automatically enables reactor trips on low flow in more than one primary coolant loop, more than one reactor coolant pump breaker open, reactor coolant pump bus undervoltage and underfrequency, ~~turbine trip,~~ pressurizer low pressure and pressurizer high level. On decreasing power the above listed trips are automatically blocked. The P-7 Interlock is a logic Function that receives input From P10 and P-13. P-7 HAS NO Parameter with which to associate an LSSS. Therefore, The Trip Setpoints And Allowable Values are not applicable to this interlock.

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

P-9 On increasing power P-9 automatically enables reactor trip on turbine trip. On decreasing power P-9 automatically blocks reactor trip on turbine trip.

P-10 On increasing power P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip. ~~and de-energizes the Source Range high voltage power.~~ On decreasing power the Intermediate Range reactor trip and the low setpoint Power Range reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range reactor trip. On decreasing power, Source Range level trips are automatically reactivated.

P-7 On increasing power P-7 automatically enables reactor trips on low flow in more than one primary coolant loop, more than one reactor coolant pump breaker open, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. On decreasing power the above listed trips are automatically blocked. The P-7 Interlock is a logic function that receives input from P-10 and P-13. P-7 has no parameter with which to associate an LSSS. Therefore, the trip setpoints and allowable values are not applicable to this interlock.

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

P-9 On increasing power P-9 automatically enables reactor trip on turbine trip. On decreasing power P-9 automatically blocks reactor trip on turbine trip.

P-10 On increasing power P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip. On decreasing power the Intermediate Range reactor trip and the low setpoint Power Range reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

TRIP

REACTOR ~~PROTECTION SYSTEM~~ AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

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

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of safety injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates safety injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of safety injection actuation on low pressurizer pressure.

P-12 On increasing primary coolant loop temperature, P-12 automatically reinstates safety injection actuation and steam line isolation on low steam line pressure, and removes a blocking signal from the steam dump system. On decreasing primary coolant loop temperature, P-12 allows the manual block of safety injection actuation and steam line isolation on low steam line pressure and automatically provides a blocking signal to the steam dump system.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

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 reactor 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.

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

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

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of safety injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates safety injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of safety injection actuation on low pressurizer pressure.

P-12 On increasing primary coolant loop temperature, P-12 automatically reinstates safety injection actuation and steam line isolation on low steam line pressure, and removes a blocking signal from the steam dump system. On decreasing primary coolant loop temperature, P-12 allows the manual block of safety injection actuation and steam line isolation on low steam line pressure and automatically provides a blocking signal to the steam dump system.

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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.

3/4.5 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 each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures. In addition, the borated water serves to limit the maximum power which may be reached during large secondary pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves ~~are~~ were originally 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. INSERT 1 In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

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

3/4.5.2 and 3/4.5.3 EMERGENCY CORE COOLING SYSTEM (ECCS) SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

BASES change for B - 3.5.1

Accumulator Isolation Valves
Operating Bypass Issue

Insert 1:

In that the subject isolation valves are normally open and do not act as active valves during any emergency condition, the requirements of IEEE 279 are not applicable for this situation in that an operating bypass function is not required for these valves.

Justification for the Bases change:

These valves are required by TS to be open in Mode 3 down to 1000 psig, Mode 2, and Mode 1 to protect the core in the event of a large break LOCA. TS requires these valves to be incapable of being closed (breaker open and locked) to assure the valve cannot inadvertently close. Since the power is removed from these valves at ~ 1000 psig and the operating bypass setpoint is P-11 (1985 psig), there is no physical way that this automatic function can perform.

3/4.5.1 ACCUMULATORS

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The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves were originally 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 that the subject isolation valves are normally open and do not act as active valves during any emergency condition, the requirements of IEEE 279 are not applicable for this situation in that an operating bypass function is not required for these valves. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

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

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With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within the reactor building in the event the steam line rupture occurs within the reactor building. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 FEEDWATER ISOLATION VALVES

The OPERABILITY of the Feedwater Isolation Valves serves to (1) limit the effects of a Steam Line rupture by minimizing the positive reactivity effects of the Reactor Coolant System Cooldown associated with the blowdown, and (2) limit the pressure rise within the reactor building in the event of a Steam Line or Feedwater Line rupture within the reactor building.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on the average impact values of the steam generator material at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within the reactor building in the event the steam line rupture occurs within the reactor building. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

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The OPERABILITY of the Feedwater Isolation Valves serves to (1) limit the effects of a Steam Line rupture by minimizing the positive reactivity effects of the Reactor Coolant System Cooldown associated with the blowdown, and (2) limit the pressure rise within the reactor building in the event of a Steam Line or Feedwater Line rupture within the reactor building.

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The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

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The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

MANIPULATOR CRANE (Continued)

and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained ~~thru~~ through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the reactor building vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the reactor building. The OPERABILITY of this system is required to restrict the release of radioactive material from the reactor building atmosphere to the environment.

3/4.9.9 and 3/4.9.10 WATER LEVEL - REACTOR VESSEL and SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 SPENT FUEL POOL VENTILATION SYSTEM

The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

MANIPULATOR CRANE (Continued)

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The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

d. Proposed tests or experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

e. Proposed changes to Technical Specifications or the Operating License.

f. 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.

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

h. Review of all REPORTABLE EVENTS.

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

j. The plant Security Plan and changes thereto.

k. The Emergency Plan and changes thereto.

l. Items which may constitute a potential nuclear safety hazard as identified during review of facility operations.

m. Investigations or analyses of special subjects as requested by the Chairman of the Nuclear Safety Review Committee.

n. The unexpected offsite release of radioactive material and the report as described in 10 CFR 50.73.

o. Changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

~~p. The plant Fire Protection Program and revisions there to.~~ Not USED

AUTHORITY

6.5.1.7 The Plant Safety Review Committee shall:

a. Recommend in writing to the General Manager, Nuclear Plant Operations, approval or disapproval of items considered under 6.5.1.6a, c, d, e, j, and k above.

b. Render determinations in writing to the General Manager, Nuclear Plant Operations, with regard to whether or not each item considered under 6.5.1.6a, c, and d above constitutes an unreviewed safety question.

c. Make recommendations in writing to the General Manager, Nuclear Plant Operations, that actions reviewed under 6.5.1.6(b) above did not constitute an unreviewed safety question.

d. Provide written notification within 24 hours to the Vice President, Nuclear Operations and the Nuclear Safety Review Committee of disagreement between the PSRC and the General Manager, Nuclear Plant Operations however, the General Manager, Nuclear Plant Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

d. Proposed tests or experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

e. Proposed changes to Technical Specifications or the Operating License.

f. 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.

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

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i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

j. The plant Security Plan and changes thereto.

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RECORDS

6.5.2.10 Records of NSRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSRC meeting shall be prepared, approved and forwarded to the Vice President, Nuclear Operations, within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice President, Nuclear Operations, within 14 days following completion of the review.

c. Audit summary reports encompassed by Section 6.5.2.8 above, shall be forwarded to the NSRC and the Vice President, Nuclear Operations. Full audits shall be forwarded 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 which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Administrative Procedures will be approved as delineated in writing by the General Manager, Nuclear Plant Operations. The General Manager, Nuclear Plant Operations will approve administrative procedures, security implementing procedures, and emergency plan implementing procedures. Temporary approval to procedures which clearly do not change the intent of the approved procedures can be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedures shall approve the change.

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the General Manager, Nuclear Plant Operations. Each such modification shall be designed as authorized by Engineering Services and shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of modifications to plant nuclear safety-related structures, systems and components shall be concurred ~~in~~ by the General Manager, Nuclear Plant Operations. with

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b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the General Manager, Nuclear Plant Operations. Each such modification shall be designed as authorized by Engineering Services and shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of modifications to plant nuclear safety-related structures, systems and components shall be concurred with by the General Manager, Nuclear Plant Operations.

e. Radioactive Effluent Controls Program (Continued)

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

4) Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50;

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases or radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual or dose commitment conforming to Appendix I to 10 CFR Part 50;

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

(a) For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin; and

(b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ;

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50;

9) Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate ~~from~~ form with half-lives greater than 8 days in gaseous effluents released to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50;

10) Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

e. Radioactive Effluent Controls Program (Continued)

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7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

   (a) For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin; and

   (b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ;

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50;

9) Limitations on the annual and quarterly doses 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 to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50;

10) Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities as specified in the NRC's approved SCE&G position on Regulatory Guide 1.88, Rev. 2, October 1976.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PSRC and the NSRC.

l. Records of the service lives of all hydraulic and mechanical snubbers defined in Section 3.7.7 including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

n. Records of analysis required by the radiological environmental monitoring program.

o. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREAS

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr* shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Health Physics personnel or individuals escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned duties, provided they otherwise comply with approved radiation protection procedures for entry into high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

* Measurement made at 30 cm (12 in.) from ~~the~~ radiation source external to the body or from any surface penetrated by the radiation.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities as specified in the NRC's approved SCE&G position on Regulatory Guide 1.88, Rev. 2, October 1976.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PSRC and the NSRC.

l. Records of the service lives of all hydraulic and mechanical snubbers defined in Section 3.7.7 including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

n. Records of analysis required by the radiological environmental monitoring program.

o. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREAS

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr * but less than 1000 mrem/hr * shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Health Physics personnel or individuals escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned duties, provided they otherwise comply with approved radiation protection procedures for entry into high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

* Measurement made at 30 cm (12 in.) from radiation sources external to the body or from any surface penetrated by the radiation.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the RWP.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr* but less than 500 rads/hr** shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the duty Shift Supervisor 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 area. The maximum allowable stay time for individuals in that area shall be established prior to entry. In lieu of the stay time specification of the RWP, direct or remote continuous surveillance (such as closed circuit TV cameras) shall be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

For individual areas accessible to personnel with radiation levels greater than 1000 mrem/hr* but less than 500 rads/hr** that are located within larger areas (such as PWR containment) where no enclosure can be reasonably constructed around the individual areas, then those areas 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 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.o. This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change (s); and

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the PSRC and approval of the General Manager, Nuclear Plant Operations.

external to the body

* Measurement made at 30 cm (12 in.) from ~~the~~ radiation source or from any surface penetrated by the radiation.

external to the body

** Measurement made at 1 meter from ~~the~~ radiation source or from any surface penetrated by the radiation.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the RWP.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr* but less than 500 rads/hr** shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the duty Shift Supervisor 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 area. The maximum allowable stay time for individuals in that area shall be established prior to entry. In lieu of the stay time specification of the RWP, direct or remote continuous surveillance (such as closed circuit TV cameras) shall be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

For individual areas accessible to personnel with radiation levels greater than 1000 mrem/hr* but less than 500 rads/hr** that are located within larger areas (such as PWR containment) where no enclosure can be reasonably constructed around the individual areas, then those areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

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2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the PSRC and approval of the General Manager, Nuclear Plant Operations.

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* Measurement made at 30 cm (12 in.) from radiation sources external to the body or from any surface penetrated by the radiation.

** Measurement made at 1 meter from radiation sources external to the body or from any surface penetrated by the radiation.

SAFETY EVALUATION FOR AN ADMINISTRATIVE CHANGE TO THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Description of Amendment Request

SCE&G proposes to change the current VCSNS Technical Specifications (TS) with an administrative change.

Several editorial changes are proposed to 3/4.8.1.1 that will provide some flexibility in how the ASTM standards are applied during surveillance testing of Emergency Diesel Generator (EDG) fuel oil. Instead of testing in accordance with the ASTM standards, we will be performing tests based on the ASTM standards. This will allow for only performing the tests that are applicable and appropriate, and allows for the inclusion of newly developed tests into our program.

A change is proposed to page 2-7 to remove the Trip Setpoints and Allowable Values associated with Reactor Trip System (RTS) interlock P-7. This change will eliminate the conflicting values provided due to the P-7 interlock receiving input from both P-10 and P-13, each with their own Trip Setpoint and Allowable Value. Since the P-7 function is a derived logic function only, there is no need to repeat the values for P-10 and P-13.

A Bases change is proposed that will correct the description for RTS Interlock P-7 on page B 2-8. This interlock lists a reactor trip on turbine trip and was inadvertently omitted from Amendment 34. The reactor trip on turbine trip function is provided by RTS interlock P-9 (50% reactor power). Another proposed change to this description provides the rationale for the removal of specific setpoint and allowable value information from Table 2.2-1.

Other proposed Bases changes to page B 2-8 involve RTS Interlocks P-6 and P-10. When the change for Amendment 65 was submitted to permit the use of the Gamma-Metrics system for the Excore Nuclear Instrumentation System, these interlocks were omitted. There is no longer an automatic deactivation of the power supplies for the Source Range detectors at the interlock setpoint.

Additionally, Bases section 3/4.5.1 "Accumulators" is being revised to address the requirement for "Operating Bypasses" in the context of IEEE Standard 279-1971 for the accumulator isolation valves. This change is the result of a Westinghouse vendor bulletin (NSAL 97-003) and will accurately reflect the plant op ting philosophy for these valves.

Section 6.5.1 is being revised per the guidance of Regulatory Information Summary (RIS) 00-02 to delete the requirement for the onsite Plant Safety Review Committee (PSRC) to review all changes to the fire protection program. This will allow the PSRC to expend its resources on more significant issues.

A proposed change for Section 6.12, High Radiation Areas, will change the footnotes that provide the definitions of high radiation area and very high radiation area. These changes are consistent with the recent change to 10 CFR 20.1003.

Additionally, multiple proposed editorial changes are included. These changes have no impact upon the plant, software, methodologies, practices, or philosophy. These changes are listed in Attachment 1 and are entirely administrative in nature.

Safety Evaluation

Allowing the proposed flexibility in applicability of the ASTM Standards pertaining to fuel oil surveillance testing will permit VCSNS to perform only those portions of those Standards that are required to assure fuel oil quality. The other reason for requesting this change is to permit equivalent testing equipment or methodologies than that specified in the applicable Standard. This change will replace the words "in accordance with" with the words "based on" to provide this flexibility. The testing that provides continued assurance of operability for the Emergency Diesel Generators will remain in the testing program. The fuel oil testing program is located in chemistry procedures and can be revised through 10 CFR 50.59.

Removing the setpoints and allowable values from Table 2.2-1, item 19.B, removes the conflicting values currently provided. Both RTS interlocks P-10 and P-13 are used to determine the P-7 interlock. Each of these interlocks receive input from instrumentation sensing different parameters and has a specific setpoint and allowable value associated. P-7 is a logic function and does not directly sense plant parameters, and as such should not have specific values listed as Limiting Safety System Setpoints. Additionally, although the setpoint for both P-10 and P-13 is 10% power (either nuclear or turbine equivalent), the allowable values permit a range in different directions. This means that the calibration of P-7 would require setting it at exactly 10% with absolutely no tolerance. This change incorporates the guidance in NUREG 1431, Revision 1.

The proposed changes to the Technical Specification Bases provide clarification and guidance to plant operators. The change that adds the discussion pertaining to the "operating bypass" function of the accumulator isolation valves clarifies the position that VCSNS has regarding these valves. This function is defeated by the removal of power from the valve while it is in the correct position; the valves themselves are repositioned while the RCS is well below the Safety Injection setpoint. This was the result of analyses that concluded that the potential and consequences for a LOCA at less than power operation conditions required availability of the SI accumulators.

The "Operating Bypass" function as described in the IEEE Standard will automatically open the Accumulator Isolation Valves at the P-11 setpoint (for VCSNS - 1985 psig) assuming the valves are closed and the power is not removed. The VCSNS TS provides added safety by opening the valves manually at approximately 1000 psig and then removing the power from the valves at the breaker. As such, there is no reason to maintain or test this function since it would never be called upon to perform. However, the test procedure continues to test this function and no plan exists to delete this testing. This Bases change is to document the fact that the "operating bypass" is not required for the Accumulator Isolation Valves due to design and operational practices.

The other Bases changes correct the descriptions of the RTS Interlocks for P-6, P-7, and P-10. Amendments 34 and 65 approved changes that impacted upon these interlocks, however, these descriptions were not revised. The interlocks are correct and are verified through periodic testing.

Amendment 34 approved the incorporation of interlock P-9 into the TS. The evaluation demonstrated that a reactor trip below 50% reactor power was unnecessary to protect the core and presented excessive challenges to the Reactor Protection System. At that time the reactor trip on turbine trip function was removed from P-7.

Amendment 65 approved the Gamma-Metrics system for use as the (Excore) Nuclear Instrumentation System. The installation of fission detectors removed the need to de-activate the detectors to protect them above their rated range. At that time the automatic de-activation of the detectors was removed from the interlock circuitry.

The Bases discussion on P-7 is also being expanded to address the reason why Trip Setpoints and Allowable Values are not necessary for this interlock. The function, P-7, is a derived logic function without actual parameter input. The inputs are P-10 and P-13, which have specified Trip Setpoints and Allowable Values. Current testing methodology requires the P-7 function occur simultaneously with either the P-10 or p-13 interlock, depending on the parameter simulated.

Section 6.5.1 discusses the Plant Safety Review Committee (PSRC), their responsibilities and duties. One of their duties is to review all changes to the Fire Protection Program to assure that the level of safety is not diminished. Per the guidance located in Regulatory Information Summary (RIS) 00-02, this requirement may be deleted from the Licensing Basis of the plant. The justification for this change is that existing regulations provide adequate assurance of continued effectiveness of the program. The required reviews will be programmatically driven allowing the PSRC to devote more time to the more significant issues and program/procedure changes. Changes to the Fire Protection program and its implementing procedures will be per 10 CFR 50.59 and will receive the appropriate reviews. Should there be an Unreviewed Safety Question, the change would have to be approved by the NRC prior to implementation. The PSRC will still review all 10 CFR 50.59 safety evaluations to assure there are no Unreviewed Safety Questions being introduced by the change.

Section 6.12, provides the plant controls for assuring that the limits of 10 CFR 20 are not exceeded. The footnotes provide clarification on the measurement of radiation associated with a high radiation area or very high radiation area. 10 CFR 20.1003 was revised which clarified the method of measuring the dose rate and the source of the radiation. This change assures the footnotes are consistent with the changes to Part 20.

Various administrative changes are being proposed that have no impact on the intent of any Technical Specification. They have been reviewed and been determined to be editorial changes only. These changes are identified in Attachment I and correct typographical errors or assure consistency throughout TS.

NO SIGNIFICANT HAZARDS EVALUATION FOR AN ADMINISTRATIVE CHANGE TO THE VIRGIL C. SUMMER NUCLEAR STATION TECHNICAL SPECIFICATIONS

Description of Amendment Request

SCE&G proposes to change the current VCSNS Technical Specifications (TS) with an administrative change.

Several editorial changes are proposed to 3/4.8.1.1 that will provide some flexibility in how the ASTM standards are applied during surveillance testing of Emergency Diesel Generator (EDG) fuel oil. Instead of testing in accordance with the ASTM standards, we will be performing tests based on the ASTM standards. This will allow for only performing the tests that are applicable and appropriate, and allows for the inclusion of newly developed tests into our program.

A change is proposed to page 2-7 to remove the Trip Setpoints and Allowable Values associated with Reactor Trip System (RTS) interlock P-7. This change will eliminate the conflicting values provided due to the P-7 interlock receiving input from both P-10 and P-13, each with their own Trip Setpoint and Allowable Value. Since the P-7 function is a derived logic function only, there is no need to repeat the values for P-10 and P-13.

A Bases change is proposed that will correct the description for RTS Interlock P-7 on page B 2-8. This interlock lists a reactor trip on turbine trip and was inadvertently omitted from Amendment 34. The reactor trip on turbine trip function is provided by RTS interlock P-9 (50% reactor power). Another proposed change to this description provides the rationale for the removal of specific setpoint and allowable value information from Table 2.2-1.

Other proposed Bases changes to page B 2-8 involve RTS Interlocks P-6 and P-10. When the change for Amendment 65 was submitted to permit the use of the Gamma-Metrics system for the Excore Nuclear Instrumentation System, these interlocks were omitted. There is no longer an automatic deactivation of the power supplies for the Source Range detectors at the interlock setpoint.

Additionally, Bases section 3/4.5.1 "Accumulators" is being revised to address the requirement for "Operating Bypasses" in the context of IEEE Standard 279-1971 for the accumulator isolation valves. This change is the result of a Westinghouse vendor bulletin (NSAL 97-003) and will accurately reflect the plant operating philosophy for these valves.

Section 6.5.1 is being revised per the guidance of Regulatory Information Summary (RIS) 00-02 to delete the requirement for the onsite Plant Safety Review Committee (PSRC) to review all changes to the fire protection program. This will allow the PSRC to expend its resources on more significant issues.

A proposed change for Section 6.12, High Radiation Areas, will change the footnotes that provide the definitions of high radiation area and very high radiation area. These changes are consistent with the recent change to 10 CFR 20.1003.

Additionally, multiple proposed editorial changes are included. These changes have no impact upon the plant, software, methodologies, practices, or philosophy. These changes are listed in Attachment 1 and are entirely administrative in nature.

Basis for No Significant Hazards Consideration Determination

In accordance with 10 CFR 50.92, a proposed change to the operating license involves no "significant hazards" if operation of the facility, in accordance with the proposed change, would not: 1) involve a significant increase in the probability or consequences of any accident previously evaluated; 2) create the possibility of a new or different kind of accident from previously evaluated, or; 3) involve a significant reduction in a margin to safety.

This request is evaluated against each of these criteria as follows:

1. *This request does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The proposed editorial change to the fuel oil surveillance requirements will permit VCSNS to apply the applicable portions of the applicable ASTM standards rather than incorporate the whole standard. The surveillance programs at VCSNS will remain sufficient to assure the continued operability of the EDGs and will not introduce any common mode failure mechanisms. The additional testing beyond what is necessary to assure operability does not provide sufficient benefit to justify verbatim compliance with these standards in their entirety. The fuel oil will continue to be tested upon receipt and at a monthly frequency for the characteristics specified in the TS, however, the testing equipment or methodology may not be identical to that specified in the standards. Equivalent testing equipment and methodologies are not capable of affecting the operability of the systems, structures or components necessary to prevent or mitigate the consequences of an accident previously evaluated.

The elimination of setpoints and allowable values for Reactor Trip System instrumentation interlock P-7 only removes the conflicting information presently found in Table 2.2-1. With both P-10 and P-13 providing input into the P-7 logic function, the respective setpoints and allowable values for these interlocks do not need to be repeated for P-7. With the range for P-10 between 10% and 12.2% RTP and range for P-13 between 10% and 7.8% turbine impulse pressure equivalent, if P-7 measured the plant parameters directly, it would be extremely difficult to calibrate P-7 accurately. There is no change proposed to any setpoint, and all calibration and testing requirements are unaffected.

Technical Specification Bases are one of the recommended resources to be utilized during a review of the station's Licensing and Design basis. Previous TS amendments have corrected the information for P-6, P-7, and P-10 in other locations but inadvertently omitted this Bases page. Correcting this information will prevent any confusion over the correct description for these Reactor Trip System Interlocks. Additionally, the added discussion for P-7 provides the justification for the removal of the Setpoints and Allowable Values for this interlock function. There is no impact on any hardware, software, training material or procedures resulting from this change

Accumulator Isolation Valves were initially classified as "operating bypasses" where they would receive a signal to open when the permissive signal was present, at and above P-11 (1985 psig). This function is defeated by both the TS requirement and the operating philosophy of the plant, which opens the valves well before the interlock setpoint and removes power to the valves in order to prevent spurious operation. In that these valves are required to be open prior to reactor start-up and do not act as active valves in any accident scenario, the requirements of IEEE 279-1971 with regards to "operating bypasses" are not applicable to these valves. Therefore, the addition of a clarification into the ECCS Bases, detailing the lack of need for this function has no impact on any installed hardware, software, or procedures. This function will continue to be tested, per plant surveillance procedure (STP-105.009), on an 18-month frequency, while the plant is shut down.

Deleting the requirement for the Plant Safety Review Committee to review all changes to the Fire Protection Program prior to implementation has no significant impact on the effectiveness of the program. This is the determination published in Regulatory Information Summary (RIS) 00-02, which concludes that the limited resources of the PSRC are better utilized in other areas. There are regulations that govern the Fire Protection Program to with specific requirements; this additional review is not required by regulation. Changes to the program will be reviewed by appropriate plant personnel; any changes that result in an Unreviewed Safety Question will require prior NRC approval. No impact to plant equipment or operating and test procedures is proposed that could affect the response to any accident scenario previously evaluated.

Revising the footnotes in Section 6.12 to provide additional clarification on the methodology used to classify radiation areas in the plant has no impact on the operation or emergency response of the plant. This change is being made to be consistent with 10 CFR 20.1003.

These changes do not adversely impact the design requirements, operating characteristics, system functions, or system interrelationships. The failure mechanisms for the accidents previously analyzed are not affected, and no additional failure modes are created. As such, this proposed change would not create any increase in probability or consequences for any previously evaluated accident.

2. *This request does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

The proposed changes to the EDG fuel oil surveillance requirements will not adversely impact the functionality of the EDGs and will continue to assure that the fuel oil procured is acceptable for immediate and prolonged use in the Emergency Diesel Generators. Since the EDGs are used to mitigate specific accident scenarios involving the loss of offsite power, a change to this surveillance requirement cannot be an accident initiator or precursor. The capability and load profile are unchanged and EDG reliability is not threatened; the performance criteria established by the maintenance rule will still be met.

The proposed elimination of specific values for Reactor Trip System Instrumentation Interlock, item 19.B (P-7) located in Table 2.2-1, will not impact plant operation or response to transients. Since P-7 is a derived logic function, which obtains input from other RTS interlocks, there is no change to the facility or calibration and testing procedures. This change clears up an apparent conflict in the Limiting Safety System Setpoints for P-7 since the two inputs into P-7 have different allowable values. This change is consistent with NUREG 1431, Revision 1.

Several Bases changes are proposed for page B 2-8, Reactor Trip System Interlocks. Interlocks P-6 and P-10 were supposed to be included in the change request that resulted in Amendment 65 but were inadvertently omitted. The Gamma-Metrics system replaced the original Nuclear Instrumentation system and eliminated any need to deactivate the Source Range detectors once over-ranged. The proposed change revises the description of the interlocks to delete the incorrect statements. This Bases change reflects approved plant configuration and operational procedures.

Another Bases change proposed for page B 2-8 pertains to Interlock P-7. The description for this interlock states that there is a reactor trip on turbine trip that is armed once permissive P-7 is met. This is incorrect and should have been corrected during the change process for Amendment 34, which incorporated Interlock P-9. Currently, the reactor will trip on a turbine trip above the P-9 setpoint (50% reactor power). Additionally, the Bases for P-7 are being expanded to provide a discussion on the reason for not providing setpoints or allowable values. These Bases changes do not have any effect on the facility or how the facility is operated; this is not a change that could affect the plant or personnel response to an event.

Bases section B 3.5.1 is being revised to address the statement that the Accumulator Isolation Valves are "operating bypasses" in the context of IEEE 279-1971. This function is defeated by TS requirements and operating philosophy by opening the breaker once the valves are in the correct position and the breaker is then locked open. These actions are performed at approximately 1000 psig, well before the P-11 setpoint of 1985 psig at which this function is to occur. This action precludes the valves from inadvertently changing position. In addition, these valves are not considered as active valves for any accident scenario, therefore there is no reason in their being considered as "operating bypass" valves. The purpose for these valves is to mitigate the effects of specific accidents; the position of these valves cannot cause any accident nor a precursor for any accident.

The proposed change to the Administrative Controls section of TS involves revising a footnote to comply with the definitions provided in a recent revision to 10 CFR 20. This change would not create any kind of accident nor alter the response of the plant or operating staff in the event of an accident.

Removing the requirement for the PSRC to review all changes to the Fire Protection Program does not affect the operating or testing methodology of the plant. The effectiveness of the program is assured through regulatory requirements, with significant changes receiving a 10 CFR 50.59 safety evaluation. These safety evaluations receive review from the PSRC to assure no USQ has occurred. The change in the review of the Fire Protection Program does not impact any equipment that could create any accident.

The proposed changes discussed above do not adversely impact any hardware, software, operating procedures or philosophy. Equipment operation and testing methodologies are not significantly altered and all safety related functions remain protected. Therefore, these proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

3. *This request does not involve a significant reduction in a margin to safety.*

The proposed editorial change to the EDG fuel oil surveillance requirements will not reduce the ability of the EDGs to perform their design function in the time frame assumed in the safety analysis. The fuel oil testing process will be consistent with the applicable ASTM standards while providing flexibility, and will assure that there will not be any reduction in load carrying capability or reliability due to unacceptable oil characteristics.

The proposed removal of the setpoint and allowable value information on Table 2.2-1, item 19.B (P-7), will not prevent the RTS interlock from performing the safety function. This interlock receives input from both P-10 and P-13, which remain unaffected by this change. The logic function P-7 is also unaffected by this change. The conflicting information provided for this function was not used in calibrating or testing this function and no facility changes are proposed.

The proposed Bases changes to page B 2-8 correct the descriptions of several of the Reactor Trip System Interlocks. Interlocks P-6 and P-10 had their circuitry revised for Amendment 65, while interlock P-7 had its circuitry modified for Amendment 34. Only the description of the function of these interlocks is being revised, the actual change to the interlocks occurred long ago. Additionally, the description of interlock P-7 is being expanded to provide the reason why there is no required setpoint or allowable value information for this function. This does not require any changes to the interlocks or any other protection cabinet components.

The proposed Bases change to Section B 3/4.5.1 clarifies the plant position on the “operating bypass” function of the Accumulator Isolation Valves. These valves are maintained in their accident-required position at all times during the modes where the accumulators are necessary. There are no changes to the design basis, hardware, software, or operating practices from this proposed change.

The proposed change to the Administrative Controls section revises footnotes in the High Radiation Area control section. This change is consistent with a recent change to 10 CFR 20.1003 and other than defining the measurement methodology, has no impact on the plant or plant procedures.

The PSRC review of the Fire Protection Program is to assure that the plant remains in a safe approved condition. Any changes to the program need to insure that the effectiveness is not reduced. This can be achieved through the existing regulatory requirements and does not require a PSRC review for every program or procedure change. As there is no impact on installed plant equipment or operating practices, there is no reduction in margin of safety.

As there are equivalent levels of quality for the fuel oil surveillances, no loss of required actions from the addition of Functional Unit 4.e to Table 4.3-2, no effect on the plant from the Bases changes or Administrative Controls section there cannot be any significant reduction in the margin to safety for any of the proposed changes.

Environmental Impact Consideration

SCE&G has reviewed this request against the criteria of 10 CFR 51.22 for environmental considerations. Since this request involves (i) no significant hazard consideration, (ii) no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (iii) no significant increase in individual or cumulative occupational radiation exposure, SCE&G has concluded that the proposed change meets the criteria given in 10 CFR 51.22 (c)(9) for a categorical exclusion from the requirement for an environmental impact statement.