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AUG 1 1 2006

LR-N06-0292
LCR S05-04 Rev. 1

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

**REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS
RELOCATION OF RESPONSE TIME TESTING TIME LIMITS
SALEM NUCLEAR GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75
DOCKET NO. 50-272 AND 50-311**

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests revision to the Technical Specifications (TS) for the Salem Nuclear Generating Station Units 1 and 2. The proposed amendment supersedes our request of August 19, 2005 (PSEG Letter LR-N05-0336, LCR S05-04), to address issues discussed with members of the NRC staff on May 4, 2006 regarding potential misapplication of the previously proposed changes to TS Definition 1.12, ENGINEERED SAFETY FEATURES RESPONSE TIME, and Definition 1.26, REACTOR TRIP SYSTEM RESPONSE TIME. This revision to our amendment request also changes the numbering of the new Updated Final Safety Analysis Report (UFSAR) tables, and identifies editorial corrections to the Unit 1 Reactor Trip Response Time and Engineered Safety Feature Response Time Table headings (i.e., changing "Response Items" to "Response Times").

PSEG proposes to revise the Salem Unit 1 and 2 TS to relocate response time limits for Reactor Trip System and Engineered Safety Features from TS tables to the Salem UFSAR. The proposed changes are consistent with NRC Generic Letter 93-08, Relocation of Technical Specification Tables of Instrument Response Time Limits, dated December 29, 1993. Beaver Valley Power Station was issued a similar amendment dated January 20, 1998 (TAC Nos. M99671 and M99672).

PSEG is also revising TS Bases Section B3/4.3 to clarify that response time acceptance criteria are relocated to the UFSAR, and plant-specific NRC approval would be required to use a means other than testing to verify response times are within limits. These Bases changes are included in lieu of the proposed changes to Definitions 1.12 and 1.26 in our August 19, 2005 request.

PSEG has evaluated the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined this request involves no significant hazards considerations. An evaluation of the requested changes is provided in Attachment 1. The marked up TS and TS Bases pages affected by the proposed

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change is provided in Attachment 2. Attachment 3 contains the UFSAR pages that will be added to the Salem UFSAR as part of implementation of the requested amendment. In accordance with 10CFR50.91(b)(1), a copy of this submittal is being sent to the State of New Jersey.

PSEG requests approval of this amendment request by June 15, 2007, with a 90 day implementation period.

Should you have any questions regarding this request, please contact Mr. James Mallon at 610-765-5507.

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

Sincerely,

Executed on 8/11/06

Thomas P. Joyce
Site Vice President
Salem Station Units 1 and 2

Attachments (3)

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**SALEM NUCLEAR GENERATING STATION UNIT NOS. 1 AND 2
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**REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS
RELOCATION OF RESPONSE TIME TESTING TIME LIMITS**

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CHANGES TO TECHNICAL SPECIFICATIONS

1.0 DESCRIPTION

PSEG requests changes to the Salem Units 1 and 2 Technical Specifications (TS). The requested changes would relocate the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) response times from TS Tables 3.3-2 and 3.3-5 to the Updated Final Safety Analysis Report (UFSAR). Neither the response time limits nor the surveillance requirements for performing response time testing would be altered by these proposed changes. Future changes to the response time limits included in the UFSAR will be controlled in accordance with the requirements of 10CFR50.59. Deletion of Response Time Testing Requirements will require prior NRC approval. In addition, changes to TS Bases Section B3/4.3 clarify that response time acceptance criteria are relocated to the UFSAR, and plant-specific NRC approval would be required to use a means other than testing to verify response times are within limits.

Editorial changes to Tables 7.3-8, Item 14 and Table 7.3-9, Item 14 are included to correct the initiating signal from Station Blackout to Loss of Offsite Power, which is the correct terminology. Editorial changes are also made to the Unit 1 Tables, 7.2-4 and 7.3-8, to correct the table headings from "Response Items" to "Response Times."

2.0 PROPOSED CHANGES

The following TS and TS Bases pages are affected by this request and the appropriate mark-ups are included in Attachment 2:

Salem Unit 1

3/4 3-1	Reactor Trip System Instrumentation
3/4 3-9	Table 3.3-2
3/4 3-10	Table 3.3-2
3/4 3-14	Engineered Safety Feature Actuation System Instrumentation
3/4 3-27	Table 3.3-5
3/4 3-28	Table 3.3-5
3/4 3-29	Table 3.3-5
3/4 3-30	Table 3.3-5
3/4 3-31	Table 3.3-5 Notation
B3/4 3-1a	Instrumentation – Bases

Salem Unit 2

3/4 3-1	Reactor Trip System Instrumentation
3/4 3-9	Table 3.3-2
3/4 3-10	Table 3.3-2
3/4 3-14	Engineered Safety Feature Actuation System Instrumentation
3/4 3-28	Table 3.3-5
3/4 3-29	Table 3.3-5
3/4 3-30	Table 3.3-5
3/4 3-31	Table 3.3-5
3/4 3-32	Table 3.3-5 Notation
B3/4 3-1a	Instrumentation – Bases

3.0 EVALUATION

The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation provides four criteria for determining whether particular limiting conditions for operation are required to be included in the TS. These are:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Existing TS limiting conditions for operation, which do not satisfy these four specified criteria, may be relocated to the UFSAR, such that future changes could be made to these provisions pursuant to 10 CFR 50.59.

NRC Generic Letter (GL) 93-08, "Relocation of Technical Specification Tables of Instrument Response Time Limits," dated December 29, 1993, provides guidance to licensees proposing to relocate RTS and ESFAS instrument response time limits from the TS to the UFSAR. GL 93-08 provides that relocation of the RTS and ESFAS instrument response time limits from the TS to the UFSAR should not alter the surveillance requirements. After relocation, the UFSAR will contain the response time limits for the RTS and ESFAS instruments, including those channels for which the response time limit is indicated as not applicable. The UFSAR also clarifies response time limits where footnotes are included in the tables that describe how those limits are applied. The limiting condition for operation (LCO) for the RTS and ESFAS instruments is modified to delete the phrases that refer to RESPONSE TIMES as shown in Table 3.3-2 (RTS) or Table 3.3-5 (ESFAS), so as to simply state that this instrumentation "shall be OPERABLE." Although the surveillance requirements for the RTS and ESFAS instrument response time limits do not reference the tables containing these limits and, therefore, do not need to be modified to implement this change, a footnote in TS Table 3.3-2 states that neutron detectors are exempt from response time testing. To retain this exception, which is stated in TS Table 3.3-2 (being removed from the TS by this proposed amendment), the RTS surveillance requirements are modified to add the following statement: "Neutron detectors are exempt from response time testing."

The change to the TS Bases B3/4.3 will clarify the need for plant-specific NRC approval of any methodology used to replace response time testing with an alternate means of verification.

The proposed changes relocate the RTS and ESFAS instrument response time limits from the TS to the UFSAR, and do not alter the surveillances for these instruments or change any of the response time limits, including those channels for which the response time limit is indicated as not applicable. The clarifications provided in the applicable TS footnotes describing how the response time limits are to be applied will also be relocated to the UFSAR. Any future changes to the RTS and ESFAS instrument response time limits will be performed in accordance with the requirements of 10 CFR 50.59. The surveillance requirements for the RTS are revised to include the footnote "Neutron detectors are exempt from response time testing," which was previously included on TS Table 3.3-2. These proposed changes are consistent with the guidance provided in GL 93-08.

4.0 REGULATORY SAFETY ANALYSIS

4.1 <u>No Significant Hazards Consideration</u>

As required by 10 CFR 50.91(a), PSEG provides its analysis of the no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated;
2. create the possibility of a new or different kind of accident from any accident previously evaluated;
3. involve a significant reduction in a margin of safety

The determinations that the criteria set forth in 10 CFR 50.92 are met for this amendment request are indicated below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment relocates the instrument response time limits for the reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) from the technical specifications to the Updated Final Safety Analysis Report (UFSAR). The proposed amendment conforms to the guidance given in Enclosures 1 and 2 of Generic Letter 93-08. Neither the response time limits nor the surveillance requirements for performing response time testing will be altered by this submittal. The overall RTS and ESFAS functional capabilities will not be changed and assurance that action requirements of the reactor trip and engineered safety features systems are completed within the time limits assumed in the accident analyses is unaffected by the proposed amendment.

   Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed amendment will not change the physical plant or the modes of plant operation defined in the operating license. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems.

   Therefore, operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The measurement of instrumentation response times at the frequencies specified in the technical specification provides assurance that actions associated with the reactor trip and engineered safety features are accomplished within the time limits assumed in the accident analyses. The response time limits and the measurement frequencies remain unchanged by the proposed amendment.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on this review, it is concluded that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, PSEG proposes that a finding of "no significant hazards consideration" is justified.

4.2 Applicable Regulatory Requirements/Criteria

The regulatory bases and guidance documents associated with the amendment request include the general design criteria that were followed in the design of the Salem Station which are the Atomic Industrial Forum (AIF) version, as published in a letter to the Atomic Energy Commission from E. A. Wiggin, Atomic Industrial Forum, dated October 2, 1967. In addition to the AIF General Design Criteria, the Salem Generating Station(SGS) was designed to comply with Public Service Electric & Gas (PSE&G's) understanding of the intent of the AEC's proposed General Design Criteria, as published for comment by the AEC in July, 1967. The application of AEC's proposed General Design Criteria to the Salem Station is discussed in UFSAR Section 3.1.2.

No changes to the RTS or ESFAS instrumentation design are requested, thus there would be no adverse impact to the General Design Criteria described above.

<u>10CFR 50.36 Technical Specifications</u>

The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation provides four criteria for determining whether particular limiting conditions for operation are required to be included in the TS.

Existing TS limiting conditions for operation which do not satisfy the four 10 CFR 50.36 criteria may be relocated to the UFSAR, such that future changes could be made to these provisions pursuant to 10 CFR 50.59.

Based on the evaluation, PSEG believes that the proposed TS changes do not reduce the level of safety maintained by the TS and are in accordance with 10 CFR 50.36.

CONCLUSION

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL ASSESSMENT/IMPACT STATEMENT

Pursuant to 10 CFR 51.22 (b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

The proposed amendment would not change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, nor change surveillance requirements. PSEG has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22 (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1 Code of Federal Regulations, General Design Criteria and 10 CFR 50.36
6.2 Salem Units 1 and 2, Updated Final Safety Analysis Report
6.3 Salem Units 1 and 2, Technical Specifications
6.4 NRC Generic Letter 93-08, Relocation of Technical Specification Tables of Instrument Response Time Limits
6.5 Beaver Valley Power Station Amendments 210 and 88 (TAC Nos M99671 AND M99672) dated January 20, 1998
6.6 NRC Information Notice 97-28, Elimination of Response Time Testing under the Requirements of 10 CFR 50.59.

REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS
RELOCATION OF RESPONSE TIME TESTING LIMITS

SALEM NUCLEAR GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75

SALEM UNIT 1 AFFECTED PAGES

3/4 3-1	Reactor Trip System Instrumentation
3/4 3-9	Table 3.3-2
3/4 3-10	Table 3.3-2
3/4 3-14	Engineered Safety Feature Actuation System Instrumentation
3/4 3-27	Table 3.3-5
3/4 3-28	Table 3.3-5
3/4 3-29	Table 3.3-5
3/4 3-30	Table 3.3-5
3/4 3-31	Table 3.3-5 Notation
B3/4 3-1a	Instrumentation Bases

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE. ~~with RESPONSE TIMES as shown in Table 3.3-2.~~ Delete

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-1.

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

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

ADD

Neutron detectors are exempt from response time testing.

DELETE.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE ITEMS

FUNCTIONAL UNIT		RESPONSE TIME
1.	Manual Reactor Trip	NOT APPLICABLE
2.	Power Range, Neutron Flux	≤ 0.5 seconds*
3.	Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4.	Power Range, Neutron Flux, High Negative Rate	≤ 0.5 seconds*
5.	Intermediate Range, Neutron Flux	NOT APPLICABLE
6.	Source Range, Neutron Flux	NOT APPLICABLE
7.	Overtemperature ΔT	≤ 5.75 seconds*
8.	Overpower ΔT	NOT APPLICABLE
9.	Pressurizer Pressure--Low	≤ 2.0 seconds
10.	Pressurizer Pressure--High	≤ 2.0 seconds
11.	Pressurizer Water Level--High	NOT APPLICABLE

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

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~~TABLE 3.3-2 (Continued)~~

~~REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES~~

FUNCTIONAL UNIT		RESPONSE TIME
12.	Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
13.	Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
14.	Steam Generator Water Level-- Low-Low	≤ 2.0 seconds
15.	Deleted	
16.	Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds
17.	Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
18.	Turbine Trip	
	A. Low Fluid Oil Pressure	NOT APPLICABLE
	B. Turbine Stop Valve	NOT APPLICABLE
19.	Safety Injection Input from ESF	NOT APPLICABLE
20.	Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE
21.	Reactor Trip Breakers	NOT APPLICABLE
22.	Automatic Trip Logic	NOT APPLICABLE

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INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

Delete

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one logic train such that both logic trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3. The provisions of Specification 4.0.4 are not applicable to MSIV closure time testing. The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump provided the surveillance is performed within 24 hours after the secondary steam generator pressure is greater than 680 psig.

DELETE

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE ITEMS

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1. Manual		
a.	Safety Injection (ECCS)	Not Applicable
	Feedwater Isolation	Not Applicable
	Reactor Trip (SI)	Not Applicable
	Containment Isolation-Phase "A"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
	Auxiliary Feedwater Pumps	Not Applicable
	Service Water System	Not Applicable
	Containment Fan Cooler	Not Applicable
b.	Containment Spray	Not Applicable
	Containment Isolation-Phase "B"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
c.	Containment Isolation-Phase "A"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
d.	Steam Line Isolation	Not Applicable
2. Containment Pressure-High		
a.	Safety Injection (ECCS)	≤27.0(1)
b.	Reactor Trip (from SI)	≤2.0
c.	Feedwater Isolation	≤10.0
d.	Containment Isolation-Phase "A"	≤17.0(2)/27.0(3)
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	≤60
g.	Service Water System	≤13.0(2)/45.0(3)
h.	Containment Fan Coolers	≤60.0 (7)

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

DELETE

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
3.	Pressurizer Pressure-Low	
a.	Safety Injection (ECCS)	$\leq$ 27.0[1]/12.0[3]
b.	Reactor Trip (from SI)	$\leq$ 2.0
c.	Feedwater Isolation	$\leq$ 10.0
d.	Containment Isolation - Phase "A"	$\leq$ 18.0[2]
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	$\leq$ 60
g.	Service Water System	$\leq$ 49.0[1]/13.0[2]
4.	Differential Pressure Between Steam Lines-High	
a.	Safety Injection (ECCS)	$\leq$ 12.0[2]/22.0[3]
b.	Reactor Trip (from SI)	$\leq$ 2.0
c.	Feedwater Isolation	$\leq$ 10.0
d.	Containment Isolation - Phase "A"	$\leq$ 17.0[2]/27.0[3]
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	$\leq$ 60
g.	Service Water System	$\leq$ 13.0[2]/48.0[3]
5.	Steam Flow in Two Steam Lines - High Coincident with T_{avg} -- Low-Low	
a.	Safety Injection (ECCS)	$\leq$ 15.75[2]/25.75[3]
b.	Reactor Trip (from SI)	$\leq$ 5.75
c.	Feedwater Isolation	$\leq$ 15.0
d.	Containment Isolation - Phase "A"	$\leq$ 20.75[2]/30.75[3]
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	$\leq$ 61.75
g.	Service Water System	$\leq$ 15.75[2]/50.75[3]
h.	Steam Line Isolation	$\leq$ 10.75

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 12.0(2)/22.0(3)
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	≤ 17.0(2)/27.0(3)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	≤ 14.(2)/48.0(3)
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 33.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 10.0
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps(4)	≤ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps(5)	≤ 60.0

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Undervoltage RCP Bus	
a. Turbine-Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Containment Radioactivity - High	
a. Purge and Pressure Vacuum Relief	≤ 5.0 (6)
12. Trip of Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
13. Undervoltage, Vital Bus	
a. Loss of Voltage	≤ 4.0
14. Station Blackout	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0

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TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(3) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(4) On 2/3 in any steam generator.

(5) On 2/3 in 2/4 steam generators.

(6) The response time is the time the isolation circuitry input reaches the isolation setpoint to the time the Isolation Valves are fully shut.

(7) The response time includes the time to automatically align the service water flow to the CFCUs following an accident coincident with a loss of offsite power, and also includes the time delays associated with isolation of the Turbine Generator Area service water header.

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No Changes to This Page - INFO Only

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as-left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in the UFSAR. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS) channels that must function in harsh environments as defined by 10 CFR 50.49, the Trip Setpoints and Allowable Values specified in the Technical Specification Limiting Conditions for Operation (LCO's) are conservatively adjusted with respect to the analytical limits. The methodology used to calculate the Trip Setpoints is consistent with Instrument Society of America standard ISA-S67.04-1982, which is endorsed via NRC Regulatory Guide 1.105, Rev. 2. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the safety analyses which demonstrate that safety limits are not violated remain valid (provided the unit is operated within the LCO's at the onset of any design basis event and the equipment functions as designed).

The Trip Setpoints and Allowable Values listed in the LCO's incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection

Response time acceptance criteria have been relocated to UFSAR Section 7.2 tables.

Instrumentation System," and Supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The verification of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety features actuation associated with each channel is completed within the time limit assumed in the safety analysis. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.).

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocation for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

, and other components that do not have plant-specific NRC approval to use alternate means of verification,

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.

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS
RELOCATION OF RESPONSE TIME TESTING LIMITS

SALEM NUCLEAR GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75

SALEM UNIT 2 AFFECTED PAGES

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REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS
RELOCATION OF RESPONSE TIME TESTING LIMITS

SALEM NUCLEAR GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75

SALEM UNIT 2 AFFECTED PAGES

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B3/4 3-1a	Instrumentation Bases

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE. ~~with RESPONSE TIMES as shown in Table 3.3-2~~ e

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

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

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

ADD

Neutron detectors are exempt from response time testing.

DELETE

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	≤ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	≤ 5.75 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	≤ 2.0 seconds
10. Pressurizer Pressure--High	≤ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

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

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TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
14. Steam Generator Water Level--Low-Low	≤ 2.0 seconds
15. Deleted	
16. Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds
17. Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE
21. Reactor Trip Breakers	NOT APPLICABLE
22. Automatic Trip Logic	NOT APPLICABLE

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INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one logic train such that both logic trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3. The provisions of Specification 4.0.4 are not applicable to MSIV closure time testing. The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump provided the surveillance is performed within 24 hours after the secondary steam generator pressure is greater than 680 psig.

DELETE

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
Containment Fan Cooler	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Ventilation Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 13.0^{(2)}/45.0^{(3)}$
h. Containment Fan Coolers	$\leq 60.0^{(7)}$

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TABLE 3.3.5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27.0[1]/12.0[2]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	≤ 18.0[2]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	≤ 49.0[1]/13.0[2]
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤ 12.0[2]/22.0[3]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation Phase "A"	≤ 17.0[2]/27.0[3]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	≤ 13.0[2]/48.0[3]
5. Steam Flow in two Steam Lines High-Coincident with T_{avg} --Low-Low	
a. Safety Injection (ECCS)	≤ 15.75[2]/25.75[3]
b. Reactor Trip (from SI)	≤ 5.75
c. Feedwater Isolation	≤ 15.0
d. Containment Isolation-Phase "A"	≤ 20.75[2]/30.75[3]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 61.75
g. Service Water System	≤ 15.75[2]/50.75[3]
h. Steam Line Isolation	≤ 10.75

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 12.0[2]/22.0[3]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	≤ 17.0[2]/27.0[3]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	≤ 14.0[2]/48.0[3]
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 33.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 10.0
9. Steam Generator Water Level --Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps(4)	≤ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps(5)	≤ 60.0

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Undervoltage RCP Bus	
a. Turbine-Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Containment Radioactivity - High	
a. Purge and Pressure Vacuum Relief	≤ 5.0(6)
12. Trip of Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
13. Undervoltage, Vital Bus	
a. Loss of Voltage	≤ 4.0
14. Station Blackout	
a. Motor Driven Auxiliary Feed Pumps	≤ 60.0
15. Semiautomatic Transfer to Recirculation	
a. ECCS valves 21SJ44, 22SJ44, 21RH4, 22RH4, 21CC16, 22CC16, 21SJ113, 22SJ113	Not Applicable

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TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(3) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(4) On 2/3 in any steam generator.

(5) On 2/3 in 2/4 steam generators.

(6) The response time is the time the isolation circuitry input reaches the isolation setpoint to the time the Isolation Valves are fully shut.

(7) The response time includes the time to automatically align the service water flow to the CFCUs following an accident coincident with a loss of offsite power, and also includes the time delays associated with isolation of the Turbine Generator Area service water header.

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3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as-left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in the UFSAR. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS) channels that must function in harsh environments as defined by 10 CFR 50.49, the Trip Setpoints and Allowable Values specified in the Technical Specification Limiting Conditions for Operation (LCO's) are conservatively adjusted with respect to the analytical limits. The methodology used to calculate the Trip Setpoints is consistent with Instrument Society of America standard ISA-S67.04-1982, which is endorsed via NRC Regulatory Guide 1.105, Rev. 2. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the safety analyses which demonstrate that safety limits are not violated remain valid (provided the unit is operated within the LCO's at the onset of any design basis event and the equipment functions as designed).

The Trip Setpoints and Allowable Values listed in the LCO's incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance

Response time acceptance criteria have been relocated to UFSAR Section 7.2 tables.

Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and Supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The verification of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety features actuation associated with each channel is completed within the time limit assumed in the safety analysis. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.).

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

, and other components that do not have plant-specific NRC approval to use alternate means of verification,

The allocation for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

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.

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS
RELOCATION OF RESPONSE TIME TESTING LIMITS

SALEM NUCLEAR GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75

UPDATED FINAL SAFETY ANALYSIS REPORT MARKED-UP CHANGES

LIST OF TABLES

Table	Title
7.2-1	List of Reactor Trips, Engineered Safety Features, Containment and Steam Line Isolation and Auxiliary Feedwater
7.2-2	Interlock Circuits
7.2-3	Legend of Analog Symbols
7.3-1	Process Instrumentation for RPS and ESF Actuation
7.3-2	Post-Accident Equipment (Inside Containment) Operational and Testing Requirements
7.3-3	Postulated Submerged Electrical Components in the Containment Following a LOCA
7.3-4	Safety Evaluation - Electrical Components and Circuits That are Affected by the Flooding of Components Within the Containment During Post-LOCA Conditions
7.3-5	Safety Evaluation - Submerged Electrical Components in the Containment During Post-LOCA Conditions
7.3-6	Safety Evaluation - Electrical Components and Circuits That are Affected by the Flooding of Components Within The Containment During Post-LOCA Conditions

INSERT

Table	Title
7.2-4	Salem Unit 1 - Reactor Trip System Instrumentation Response Times
7.2-5	Salem Unit 2 - Reactor Trip System Instrumentation Response Times

LIST OF TABLES (Cont)

Table	Title
7.3-7	Safety Evaluation - Electrical Components and Circuits That are Affected by the Flooding of Components Within the Containment During Post-LOCA Conditions - Junction/Terminal Boxes
INSERT	
7.5-1	Main Control Room Indicators and/or Recorders Available to the Operator
7.5-2	Main Control Room Indicators and/or Recorders Available to the Operator to Monitor Significant Plant Parameters During Normal Operation
7.5-3	Index Type "A" Variables
7.5-4	Summary of Instrumentation Compliance with Regulatory Guide 1.97
7.5-5	Justification for Nonconformance to Regulatory Guide 1.97
7.7-1	Rod Stops
7.7-2	Overhead Annunciator Groupings
7.7-3	Seismic Monitoring Instrumentation
7.7-4	Seismic Monitoring Instrumentation Surveillance Requirements
7.7-5	Meteorological Monitoring Instrumentation
7.7-6	Meteorological Monitoring Instrumentation Surveillance Requirements
7.10-1	Safety Parameter Display System Parameters
7.10-2	Comparisons of Safety Functions with NUREG-0737 Supplement 1
7.10-3	Critical Safety Functions Associated with Barriers
7.3-8	Salem Unit 1 - Engineered Safety Features Response Times
7.3-9	Salem Unit 2 - Engineered Safety Features Response Times

TABLE 7.2-4

SALEM UNIT 1

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE ~~ITEMS~~ TIMES

FUNCTIONAL UNIT		RESPONSE TIME
1.	Manual Reactor Trip	NOT APPLICABLE
2.	Power Range, Neutron Flux	≤ 0.5 seconds*
3.	Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4.	Power Range, Neutron Flux, High Negative Rate	≤ 0.5 seconds*
5.	Intermediate Range, Neutron Flux	NOT APPLICABLE
6.	Source Range, Neutron Flux	NOT APPLICABLE
7.	Overtemperature ΔT	≤ 5.75 seconds*
8.	Overpower ΔT	NOT APPLICABLE
9.	Pressurizer Pressure--Low	≤ 2.0 seconds
10.	Pressurizer Pressure--High	≤ 2.0 seconds
11.	Pressurizer Water Level--High	NOT APPLICABLE

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

TABLE 7.2-4

SALEM UNIT 1

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIME
12.	Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
13.	Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
14.	Steam Generator Water Level-- Low-Low	≤ 2.0 seconds
15.	Deleted	
16.	Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds
17.	Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
18.	Turbine Trip	
	A. Low Fluid Oil Pressure	NOT APPLICABLE
	B. Turbine Stop Valve	NOT APPLICABLE
19.	Safety Injection Input from ESF	NOT APPLICABLE
20.	Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE
21.	Reactor Trip Breakers	NOT APPLICABLE
22.	Automatic Trip Logic	NOT APPLICABLE

TABLE 7.2-5

SALEM UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIME
1.	Manual Reactor Trip	NOT APPLICABLE
2.	Power Range, Neutron Flux	≤ 0.5 seconds*
3.	Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4.	Power Range, Neutron Flux, High Negative Rate	≤ 0.5 seconds*
5.	Intermediate Range, Neutron Flux	NOT APPLICABLE
6.	Source Range, Neutron Flux	NOT APPLICABLE
7.	Overtemperature ΔT	≤ 5.75 seconds*
8.	Overpower ΔT	NOT APPLICABLE
9.	Pressurizer Pressure--Low	≤ 2.0 seconds
10.	Pressurizer Pressure--High	≤ 2.0 seconds
11.	Pressurizer Water Level--High	NOT APPLICABLE

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

TABLE 7.2-5

SALEM UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIME
12.	Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
13.	Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
14.	Steam Generator Water Level-- Low-Low	≤ 2.0 seconds
15.	Deleted	
16.	Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds
17.	Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
18.	Turbine Trip	
	A. Low Fluid Oil Pressure	NOT APPLICABLE
	B. Turbine Stop Valve	NOT APPLICABLE
19.	Safety Injection Input from ESF	NOT APPLICABLE
20.	Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE
21.	Reactor Trip Breakers	NOT APPLICABLE
22.	Automatic Trip Logic	NOT APPLICABLE

TABLE 7.3-8

SALEM UNIT 1

ENGINEERED SAFETY FEATURES RESPONSE ~~ITEMS~~ TIMES

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
1.	Manual		
	a.	Safety Injection (ECCS)	Not Applicable
		Feedwater Isolation	Not Applicable
		Reactor Trip (SI)	Not Applicable
		Containment Isolation-Phase "A"	Not Applicable
		Containment Ventilation Isolation	Not Applicable
		Auxiliary Feedwater Pumps	Not Applicable
		Service Water System	Not Applicable
		Containment Fan Cooler	Not Applicable
	b.	Containment Spray	Not Applicable
		Containment Isolation-Phase "B"	Not Applicable
		Containment Ventilation Isolation	Not Applicable
	c.	Containment Isolation-Phase "A"	Not Applicable
		Containment Ventilation Isolation	Not Applicable
	d.	Steam Line Isolation	Not Applicable

TABLE 7.3-8

SALEM UNIT 1

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
2.	Containment Pressure-High	
a.	Safety Injection (ECCS)	≤27.0 (1)
b.	Reactor Trip (from SI)	≤2.0
c.	Feedwater Isolation	≤10.0
d.	Containment Isolation-Phase "A"	≤17.0 (2) /27.0 (3)
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	≤60
g.	Service Water System	≤13.0 (2) /45.0 (3)
h.	Containment Fan Coolers	≤60.0 (7)
3.	Pressurizer Pressure-Low	
a.	Safety Injection (ECCS)	≤ 27.0 (1) /12.0 (2)
b.	Reactor Trip (from SI)	≤ 2.0
c.	Feedwater Isolation	≤ 10.0
d.	Containment Isolation - Phase "A"	≤ 18.0 (2)
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	≤ 60
g.	Service Water System	≤ 49.0 (1) /13.0 (2)

TABLE 7.3-8

SALEM UNIT 1

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 10.0
d. Containment Isolation - Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Service Water System	$\leq 13.0^{(2)}/48.0^{(3)}$
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg} -- Low-Low	
a. Safety Injection (ECCS)	$\leq 15.75^{(2)}/25.75^{(3)}$
b. Reactor Trip (from SI)	$\leq$ 5.75
c. Feedwater Isolation	$\leq$ 15.0
d. Containment Isolation - Phase "A"	$\leq 20.75^{(2)}/30.75^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 61.75
g. Service Water System	$\leq 15.75^{(2)}/50.75^{(3)}$
h. Steam Line Isolation	$\leq$ 10.75

TABLE 7.3-8

SALEM UNIT 1

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 14.0^{(2)}/48.0^{(3)}$
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 33.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
8. Steam Generator Water Level--High High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 10.0

TABLE 7.3-8

SALEM UNIT 1

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps(4)	≤ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps(5)	≤ 60.0
10. Undervoltage RCP Bus	
a. Turbine-Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Containment Radioactivity - High	
a. Purge and Pressure Vacuum Relief	≤ 5.0 (6)
12. Trip of Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
13. Undervoltage, Vital Bus	
a. Loss of Voltage	≤ 4.0
14. ~~Station Blackout~~ Loss of Offsite Power	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0

TABLE 7.3-8

SALEM UNIT 1

ENGINEERED SAFETY FEATURES RESPONSE TIMES

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(3) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(4) On 2/3 in any steam generator.

(5) On 2/3 in 2/4 steam generators.

(6) The response time is the time the isolation circuitry input reaches the isolation setpoint to the time the Isolation Valves are fully shut.

(7) The response time includes the time to automatically align the service water flow to the CFCUs following an accident coincident with a loss of offsite power, and also includes the time delays associated with isolation of the Turbine Generator Area service water header.

TABLE 7.3-9

SALEM UNIT 2

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1. Manual		
a.	Safety Injection (ECCS)	Not Applicable
	Feedwater Isolation	Not Applicable
	Reactor Trip (SI)	Not Applicable
	Containment Isolation-Phase "A"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
	Auxiliary Feedwater Pumps	Not Applicable
	Service Water System	Not Applicable
	Containment Fan Cooler	Not Applicable
b.	Containment Spray	Not Applicable
	Containment Isolation-Phase "B"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
c.	Containment Isolation-Phase "A"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
d.	Steam Line Isolation	Not Applicable

SALEM UNIT 2

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Containment Pressure-High	
a. Safety Injection (ECCS)	≤27.0(1)
b. Reactor Trip (from SI)	≤2.0
c. Feedwater Isolation	≤10.0
d. Containment Isolation-Phase "A"	≤17.0(2)/27.0(3)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤60
g. Service Water System	≤13.0(2)/45.0(3)
h. Containment Fan Coolers	≤60.0(7)
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27.0(1)/12.0(2)
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation - Phase "A"	≤ 18.0(2)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	≤ 49.0(1)/13.0(2)

TABLE 7.3-9

SALEM UNIT 2

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤ 12.0[2]/22.0[3]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation - Phase "A"	≤ 17.0[2]/27.0[3]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	≤ 13.0[2]/48.0[3]
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg} -- Low-Low	
a. Safety Injection (ECCS)	≤ 15.75[2]/25.75[3]
b. Reactor Trip (from SI)	≤ 5.75
c. Feedwater Isolation	≤ 15.0
d. Containment Isolation - Phase "A"	≤ 20.75[2]/30.75[3]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 61.75
g. Service Water System	≤ 15.75[2]/50.75[3]
h. Steam Line Isolation	≤ 10.75

TABLE 7.3-9

SALEM UNIT 2

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 14.0^{(2)}/48.0^{(3)}$
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 33.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
8. Steam Generator Water Level--High High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 10.0

TABLE 7.3-9

SALEM UNIT 2

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps(4)	≤ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps(5)	≤ 60.0
10. Undervoltage RCP Bus	
a. Turbine-Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Containment Radioactivity - High	
a. Purge and Pressure Vacuum Relief	≤ 5.0(6)
12. Trip of Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
13. Undervoltage, Vital Bus	
a. Loss of Voltage	≤ 4.0
14. ~~Station Blackout~~ Loss of Offsite Power	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
15. Semiautomatic Transfer to Recirculation	
a. ECCS valves 21SJ44, 22SJ44, 21RH4, 22RH4, 21CC16, 21SJ113, 22SJ113	Not Applicable

TABLE 7.3-9

SALEM UNIT 2

ENGINEERED SAFETY FEATURES RESPONSE TIMES

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(3) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(4) On 2/3 in any steam generator.

(5) On 2/3 in 2/4 steam generators.

(6) The response time is the time the isolation circuitry input reaches the isolation setpoint to the time the Isolation Valves are fully shut.

(7) The response time includes the time to automatically align the service water flow to the CFCUs following an accident coincident with a loss of offsite power, and also includes the time delays associated with isolation of the Turbine Generator Area service water header.