PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

LR-N05-0336
LCR S05-04

PSEG
Nuclear LLC

AUG 1 9 2005

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

**REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS
RELOCATION OF RESPONSE TIME TESTING TIME LIMITS
TO THE UPDATED FINAL SAFETY ANALYSIS REPORT
SALEM NUCLEAR GENERATING STATION UNITS 1 and 2
FACILITY OPERATING LICENSES DPR-70 and DPR-75
DOCKET NOs. 50-272 and 50-311**

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to the Technical Specifications (TS) for the Salem Nuclear Generating Station, Units 1 and 2. In accordance with 10 CFR 50.91 (b)(1), a copy of this submittal has been sent to the State of New Jersey.

PSEG Nuclear proposes to revise the Salem Unit 1 and 2 Technical Specifications to reflect the relocation of Response Time Testing Tables to the Updated Final Safety Analysis report in accordance with NRC guidance provided in 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 the Definition of Engineered Safety Feature Response Time and Reactor Trip System Response Time using the guidance of Improved Technical Specifications (NUREG 1431), as modified by TSTF-111 and NRC Information Notice 97-28, Elimination of Instrument Response Time Testing under the Requirements of 10 CFR 50.59, dated May 30, 1997.

PSEG has evaluated the proposed changes in accordance with 10 CFR 50.91 (a)(1), using the criteria in 10 CFR 50.92 (c), and has determined this request involves no significant hazard considerations. This amendment to the Salem TS meets the criteria of 10 CFR 51.22 (c)(9) for categorical exclusion from an environmental impact statement.

The requested changes are provided in Attachment 1 to this letter. The proposed marked up Technical Specification pages are provided in Attachment 2. Attachment 3 contains the UFSAR pages, as they will appear in the Salem UFSAR following approval of this request.

A053

Should you have any questions regarding this request, please contact Steve Mannon at 856-339-1129.

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

Sincerely,

Executed on 8/19/05

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

Attachments (3)

cc Mr. S. J. Collins, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U. S. Nuclear Regulatory Commission
ATTN: Mr. Stewart Bailey, Project Manager – Salem and Hope Creek Stations
Mail Stop 08B1
Washington, DC 20555-0001

USNRC Senior Resident Inspector - Salem (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
PO Box 415
Trenton, NJ 08625

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS RELOCATION OF RESPONSE TIME TESTING TIME LIMITS TO THE UPDATED FINAL SAFETY ANALYSIS REPORT

SALEM NUCLEAR GENERATING STATION UNITS 1 & 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOs. 50-272 AND 50-311

ATTACHMENT 1

DESCRIPTION AND EVALUATION OF REQUESTED CHANGES

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS RELOCATION OF RESPONSE TIME TESTING TIME LIMITS TO THE UPDATED FINAL SAFETY ANALYSIS REPORT

Table of Contents

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS RELOCATION OF RESPONSE TIME TESTING TIME LIMITS TO THE UPDATED FINAL SAFETY ANALYSIS REPORT

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 Feature 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 10 CFR 50.59. Deletion of Response Time Testing Requirements will require prior NRC approval. In addition, a change to the Definition of Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS), will require prior NRC review and approval of any methodology used to verify the response times for selected components in lieu of measuring them.

Editorial changes to Tables 7.2-5, Item 14 and Table 7.2-5A, Item 14, are included to correct the Initiating Signal from Station Blackout to Loss of Offsite Power, which is the correct terminology.

2.0 PROPOSED CHANGES

The following TS pages are affected by this request and the appropriate markups are included in Attachment 2:

Salem Unit 1

1-3	Definition 1.12
1-6	Definition 1.26
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 Notations

Salem Unit 2

1-3	Definition 1.12
1-6	Definition 1.26
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 Notations

No changes to the applicable Bases are proposed in accordance with the guidance of Generic Letter 93-08 and Improved Standard Technical Specifications.

Attachment 3 contains the UFSAR pages representing the proposed relocation of the Response Time Limits Tables.

Table 7.2.4, Salem Unit 1 Reactor Trip System Instrumentation Response Times
Table 7.2.5, Salem Unit 1 Engineered Safety Features Response Times
Table 7.2.4A, Salem Unit 2 Reactor Trip System Instrumentation Response Times
Table 7.2.5A, Salem Unit 2 Engineered Safety Features Response Times
Page 7.3-26
Chapter 7, Tables, Page 7-vi

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 re-

sponse 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 "NA"; that is, a response time is 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 "with RESPONSE TIMES as shown in Table 3.3-2 (RTS) or 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 on 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 amendment), the RTS surveillance requirements is modified to add the following statement: "Neutron detectors are exempt from response time testing."

The change to the Definition of Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS), will require prior NRC review and approval of any methodology used to verify the response times for selected components in lieu of measuring them. The additional wording proposed is included in the Improved Standard Technical Specifications and was incorporated as part of TSTF-111.

The proposed changes relocate the RTS and ESFAS instrument response time limits from the TS to the UFSAR, which 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 NA. 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 proposed changes also delete from the LCO, the phrase "with response TIMES as shown in Table 3.3-2 (RTS) or 3.3-5 ESFAS)" so as to simply state "shall be OPERABLE". 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 Basis for proposed no significant hazards consideration determination

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 previously evaluated; or
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. 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 accident previously evaluated.

3. Does the 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 systems 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 the 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.

10CFR 50.36 Technical Specifications

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 currently maintained by the TS and are in accordance with 10 CFR 50.36.

CONCLUSION

PSEG has concluded, based on the considerations discussed above, that: (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 amendments 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 does 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 and change surveillance requirements. PSEG has determined that the proposed amendment involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, the amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

Therefore, it has been determined that there is:

1. No significant hazards consideration,
2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and
3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, this amendment request to the Salem Technical Specifications meets the criteria of 10 CFR 51.22 (c)(9) for categorical exclusion from an environmental impact statement.

6.0 REFERENCES

6.1 Code of Federal Regulations, General Design Criteria and 10 CFR 50.36.
6.2 PSEG Salem Units 1 and 2, Updated Final Safety Analysis Report.
6.3 PSEG 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.
6.7 NUREG 1431, TSTF-111, Revise Bases for SRs 3.3.1.16 and 3.3.2.10 to eliminate pressure sensor response time testing.

EVALUATION OF REVISIONS TO THE
TECHNICAL SPECIFICATIONS
RELOCATION OF RESPONSE TIME TESTING TIME LIMITS
TO THE UPDATED FINAL SAFETY ANALYSIS REPORT

SALEM NUCLEAR GENERATING STATION UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOs. 50-272 AND 50-311

ATTACHMENT 2

TECHNICAL SPECIFICATIONS
MARKED-UP CHANGES

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS RELOCATION OF RESPONSE TIME TESTING TIME LIMITS TO THE UPDATED FINAL SAFETY ANALYSIS REPORT

Attachment 2

<u>**Salem Unit 1 Affected Pages**</u>

1-3	Definition 1.12
1-6	Definition 1.26
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 Notations

thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

ADD

In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FULLY WITHDRAWN

1.13a FULLY WITHDRAWN shall be the condition where control and/or shutdown banks are at a position which is within the interval of 222 to 228 steps withdrawn, inclusive. FULLY WITHDRAWN will be specified in the current reload analysis.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except Reactor Coolant Pump Seal Water Injection) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

ADD

In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

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

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee, as shown in Figure 5.1-3, and which defines the exclusion area as shown in Figure 5.1-1.

SOLIDIFICATION

1.30 Not Used

SOURCE CHECK

1.31 SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for (n) systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into (n) equal subintervals.

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.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE ITEMS

DELETE

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.

PAGE LEFT BLANK INTENTIONALLY

Delete

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

Page left blank intentionally

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)

PAGE LEFT BLANK INTENTIONALLY

DELETE

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

PAGE LEFT BLANK INTENTIONALLY

DELETE

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

PAGE LEFT BLANK INTENTIONALLY

DELETE

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

PAGE LEFT BLANK INTENTIONALLY

DELETE

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.

PAGE LEFT BLANK INTENTIONALLY

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS RELOCATION OF RESPONSE TIME TESTING TIME LIMITS TO THE UPDATED FINAL SAFETY ANALYSIS REPORT

Attachment 2

Salem Unit 2 Affected Pages

1-3	Definition 1.12
1-6	Definition 1.26
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 Notations

thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

ADD

In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FULLY WITHDRAWN

1.13a FULLY WITHDRAWN shall be the condition where control and/or shutdown banks are at a position which is within the interval of 222 to 228 steps withdrawn, inclusive. FULLY WITHDRAWN will be established by the current reload analysis.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except Reactor Coolant Pump Seal Water Injection) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

ADD

In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

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

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee, as shown in Figure 5.1-3, and which defines the exclusion area as shown in Figure 5.1-1.

SOLIDIFICATION

1.30 Not Used

SOURCE CHECK

1.31 SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for (n) systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into (n) equal subintervals.

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.

PAGE LEFT BLANK INTENTIONALLY

DELETE

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

PAGE LEFT BLANK INTENTIONALLY

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	$\leq 60.0^{(7)}$

PAGE LEFT BLANK INTENTIONALLY

DELETE

TABLE 3.3.5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/12.0^{(2)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{(2)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 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)	≤ 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	$\leq 13.0^{(2)}/48.0^{(3)}$
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)	≤ 5.75
c. Feedwater Isolation	≤ 15.0
d. Containment Isolation-Phase "A"	$\leq 20.75^{(2)}/30.75^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 61.75
g. Service Water System	$\leq 15.75^{(2)}/50.75^{(3)}$
h. Steam Line Isolation	≤ 10.75

PAGE LEFT BLANK INTENTIONALLY

DELETE

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)	$\leq 12.0^{(2)}/22.0^{(3)}$
	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
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

PAGE LEFT BLANK INTENTIONALLY

DELETE

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

PAGE LEFT BLANK INTENTIONALLY

DELETE

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.

PAGE LEFT BLANK INTENTIONALLY

EVALUATION OF REVISIONS TO THE
TECHNICAL SPECIFICATIONS
RELOCATION OF RESPONSE TIME TESTING TIME LIMITS
TO THE UPDATED FINAL SAFETY ANALYSIS REPORT

SALEM NUCLEAR GENERATING STATION UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOs. 50-272 AND 50-311

ATTACHMENT 3

UPDATED FINAL SAFETY ANALYSIS REPORT
MARKED-UP CHANGES

TABLE 7.2-4
SALEM UNIT-1
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 7.2-4(Continued)
SALEM UNIT-1
REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIME
12.	Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
13.	Loss of Flow - Two Loops (Above P-7 and below P-8)	$\leq$ 1.0 seconds
14.	Steam Generator Water Level-- Low-Low	$\leq$ 2.0 seconds
15.	Deleted	
16.	Undervoltage-Reactor Coolant Pumps	$\leq$ 1.2 seconds
17.	Underfrequency-Reactor Coolant Pumps	$\leq$ 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-1
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME
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 7.2-5 (Continued)
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 7.2-5 (Continued)
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)	≤ 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 Isol.	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 7.2-5 (Continued)
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)	≤ 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

TABLE 7.2-5 (Continued)
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 7.2-5 (Continued)
SALEM UNIT-1
TABLE NOTATIONS

(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 7.2.4A
SALEM UNIT-2
REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 5.75 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 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 7.2-4A (Continued)
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 7.2-5A
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

TABLE 7.2.5A (Continued)
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]

TABLE 7.2-5A (Continued)
SALEM UNIT-2
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/12.0^{(2)}$
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

TABLE 7.2-5A (Continued)
SALEM UNIT-2
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
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 7.2-5A (Continued)
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)	$\leq$ 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	$\leq 14.0^{(2)}/48.0^{(3)}$
h. Steam Line Isolation	$\leq$ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 33.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	$\leq$ 7.0

TABLE 7.2-5A (Continued)
SALEM UNIT-2
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
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
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 Feed Pumps	≤ 60.0
15. Semi-automatic Transfer to Recirculation	
a. ECCS valves 21SJ44, 22SJ44, 21RH4, 22RH4, 21CC16, 22CC16, 21SJ113, 22SJ113	Not Applicable

TABLE 7.2-5A (Continued)
SALEM UNIT-2
TABLE NOTATIONS

(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.

The method described provides capability for checking from the process signal to the logic cabinets and from there to the individual field equipment including all field cabling actually used in the circuitry. For those devices whose operation could have an effect on plant stability, the procedure provides for checking from the process signal to the logic rack and continuity determination for output cables and field devices; however, the actuated equipment will be manually initiated as plant conditions permit.

The SEC units have the following test capability during power operation:

1. Check the operational capability of each bus undervoltage sensor and its input to the logic.

2. Check the operational capability of the LOCA signal, "S", from the SSPS logics.

3. Check that the logic combinations of input signals result in proper operation of the various functions, including automatic load sequencing, without actuation of any motors and a verification of the timed loading sequence.

4. Check the output relay capability to actuate the driven equipment.

The SEC units can also be checked for complete system operability from sensor to actuated equipment during plant shutdowns.

Reactor Trip System and ESF actuation system response time tests are required by and will be performed in accordance with the Technical Specifications. **The Technical Specifications Tables containing the response time limits were relocated to UFSAR Tables 7.2.4 and 7.2.5 for Unit 1 and 7.2.4A and 7.2.5A for Unit 2. The relocation of these tables to this document was approved by the NRC in Amendments ___ for Unit 1 and ___ for Unit 2.**

LIST OF TABLES