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

MAY 01 2006

LR-N06-0035
LCR S04-09

PSEG
Nuclear LLC

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
ELIMINATION OF POWER RANGE, NEUTRON FLUX, HIGH NEGATIVE RATE TRIP AND CORRECTION OF ERRORS
SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

In accordance with the requirements of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby transmits a request for amendment of the Technical Specifications (TS) for Salem Generating Station (Salem), Unit Nos. 1 and 2. Pursuant to the requirements of 10 CFR 50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed changes involve the elimination of the requirement for a Power Range, Neutron Flux, High Negative Rate trip. The references to this trip as Functional Unit 4 in Salem Unit Nos. 1 and 2 TS Table 2.2-1,"Reactor Trip System Instrumentation Trip Setpoints," TS Table 3.3-1, "Reactor Trip System Instrumentation," TS Table 3.3-2, "Reactor Trip System Instrumentation Response Times," and TS Table 4.3-1, "Reactor Trip System Instrumentation Surveillance Requirements," are proposed for deletion. The proposed changes are consistent with the methodology presented in Westinghouse Topical Report, WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990. This report was accepted by the NRC in Letter from A. C. Thadani (U.S. NRC) to R. A. Newton (Westinghouse Owners Group), "Acceptance for Referencing of Licensing Topical Reports WCAP-11394(P) and WCAP-11395(NP), "Methodology for the Analysis of the Dropped Rod Event," dated October 23, 1989.

Other proposed changes involve the correction of errata in the TS for Salem Unit Nos. 1 and 2.

Attachment 1 provides an evaluation of the proposed changes. Attachment 2 provides the existing TS pages marked-up to show the proposed changes. Attachment 3 summarizes the regulatory commitments made in this submittal. For your information, Attachment 4 provides the existing TS Bases pages marked-up to reflect the associated changes to the TS.

A001

PSEG does not have specific schedule needs for the proposed changes and processing can be pursued in accordance with the normal NRC review schedule for this type of request. PSEG requests implementation within 60 days of receipt of the approved amendment.

If you have any questions concerning this request, please contact Mr. Paul Duke at (856) 339-1466.

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

Executed on 5/1/06
(Date)

Sincerely,

Thomas P. Joyce
Site Vice President
Salem Generating Station

Attachments (4)

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

Mr. S. Bailey, Project Manager - Salem Unit 1 and Unit 2
U. S. Nuclear Regulatory Commission
Mail Stop 08B1
Washington, DC 20555-0001

USNRC Senior Resident Inspector – Salem Unit 1 and Unit 2 (X24)

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

SALEM GENERATING STATION – UNIT 1 AND UNIT 2
FACILITY OPERATING LICENSES NOS. DPR-70 AND DPR-75
DOCKET NO. 50-272 AND 50-311

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
ELIMINATION OF POWER RANGE, NEUTRON FLUX, HIGH NEGATIVE RATE TRIP

Table of Contents

1.0 DESCRIPTION 1

2.0 PROPOSED CHANGE 2

3.0 BACKGROUND 2

4.0 TECHNICAL ANALYSIS 4

5.0 REGULATORY SAFETY ANALYSIS 6

5.1 No Significant Hazards Consideration 6

5.2 Applicable Regulatory Requirements/Criteria 8

6.0 ENVIRONMENTAL CONSIDERATION 9

7.0 REFERENCES 9

CHANGE TO TECHNICAL SPECIFICATIONS

1.0 DESCRIPTION

1.1 Elimination of the Requirement for a Power Range Neutron Flux, High Negative Rate Trip

The purpose of this amendment is to eliminate the requirement for a Power Range Neutron Flux, High Negative Rate trip. The references to this trip as Functional Unit 4 in Salem Generating Station (Salem) Units 1 and 2 Technical Specifications (TS) Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," TS Table 3.3-1, "Reactor Trip System Instrumentation," TS Table 3.3-2, "Reactor Trip System Instrumentation Response Times," and TS Table 4.3-1, "Reactor Trip System Instrumentation Surveillance Requirements," are proposed for deletion. The proposed changes are consistent with the methodology presented in the Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," which has been reviewed by the NRC and found acceptable for referencing in license applications.

1.2 Correction of TS Errors

Other proposed changes involve the correction of errors in the TS for Salem Units 1 and 2.

- An editorial change is proposed for Salem Unit 1 Index page XII to reflect the location of TS Bases Section 3/4.4.6 as page B 3/4 4-4a.

- Editorial changes are proposed for Salem Unit 1 TS Surveillance Requirement (SR) 4.2.2.2 to correct the subscript and superscript for the peaking factor term. Standard convention has the subscript followed by the superscript. The subscript and superscript are reversed in some cases in SR 4.2.2.2.

- Editorial changes are proposed for Salem Unit 2 TS SR 4.2.2.2 to correct typographical errors and eliminate a condition approved only for Cycle 11, which has expired. The subscript and superscript convention for the peaking factor term is corrected, "CORL" is corrected to "COLR" in TS SR 4.2.2.2.c.2, and the footnote for TS SR 4.2.2.2.a and TS SR 4.2.2.2.b is deleted.

- An editorial change is proposed for Salem Unit 1 TS Table 3.3-2 title to correct a typographical error. The last word in the title is changed from "items" to "times".

- An editorial change is proposed to correct a typographical error for the Salem Unit 2 Surveillance Requirement (SR) for boron concentration on page 3/4 9-1. The SR number is to be changed from 4.9.2 to 4.9.1.

- An editorial change is proposed for Salem Unit 1 TS 6.9.1.5.a to correct a typographical error and an administrative change is proposed to eliminate Salem Unit 1 TS 6.9.1.5.b. For TS 6.9.1.5.a, "totalling" is changed to "totaling." The TS 6.9.1.5.b requirement to submit results of steam generator tube inservice inspections (reference Specification 4.4.5.5.b) was superceded by the changes made in Amendment 268. Amendment 268 eliminated Specification 4.4.5.5.b and added TS 6.9.1.10, "Steam Generator Tube Inspection Report."

2.0 PROPOSED CHANGE

2.1 Elimination of the Requirement for a Power Range Neutron Flux, High Negative Rate Trip

The proposed changes revise Salem Units 1 and 2 TS Tables 2.2-1, 3.3-1, 3.3-2, and 4.3-1 to eliminate the requirement for Functional Unit 4, "Power Range, Neutron Flux, High Negative Rate."

2.2 Correction of TS Errors

Salem Unit 1 Index page XII is revised to correct the location of TS Bases Section 3/4.4.6. Salem Unit 1 TS 4.2.2.2 is revised to correct typographical errors. Salem Unit 2 TS 4.2.2.2 is revised to correct typographical errors and eliminate a condition approved only for Cycle 11, which has expired. Salem Unit 1 TS Table 3.3-2 title is revised to correct a typographical error. The Salem Unit 2 SR number for boron concentration on page 3/4 9-1 is revised from 4.9.2 to 4.9.1. Salem Unit 1 TS 6.9.1.5.a is revised to correct a typographical error. Salem Unit 1 TS 6.9.1.5.b is deleted.

3.0 BACKGROUND

3.1 Elimination of the Requirement for a Power Range Neutron Flux, High Negative Rate Trip

The dropped rod event is a Condition II event, "Fault of Moderate Frequency," that is analyzed in Salem Updated Final Safety Analysis Report (UFSAR) Section 15.2.3, "Rod Cluster Control Assembly Misalignment." The dropped rod event is assumed to be initiated by a single electrical or mechanical failure, which causes any number and combination of rods from the same group of a given bank to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to quickly decrease. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition.

If a dropped rod event occurs while in the automatic rod control mode, the Rod Control System receives signals from the excore detectors and the turbine to indicate a primary/secondary side power mismatch. In an attempt to eliminate a mismatch, the Rod Control System initiates the movement of a partially inserted control bank. Power overshoot may occur, after which the control system will insert the control bank and return the plant to nominal power. In addition to a power overshoot, an increase in the hot channel factor due to the skewed power distribution representative of a dropped rod configuration may occur.

The Power Range, Neutron Flux, Negative Rate trip channels were originally designed to provide trip signals when needed to ensure that the minimum departure from nucleate boiling ratio (DNBR) was maintained above the safety limit for all single or multiple rod drop events. As discussed in Section 4.0 below, Westinghouse Topical Report WCAP-11394-P shows that the results of the dropped rod event were acceptable with no credit taken for any direct reactor trip due to dropped rods and with no automatic power reduction features actuated by the dropped rods. Thus, the Power Range, Neutron Flux, High Negative Rate trip could be removed.

The NRC has approved a similar license amendment for the Callaway Plant, Unit 1 when it issued Amendment No. 56 on August 23, 1990 (TAC No. 76433). The NRC has also approved a similar license amendment for the South Texas Project, Units 1 and 2 when it issued Amendment Nos. 34 and 25 on March 12, 1992 (TAC Nos. M80240 and M80241).

3.2 Correction of TS Errors

Salem Unit 1 Amendment 268 approved the addition of TS Bases for Section 3/4.4.5, "Steam Generator (SG) Tube Integrity." The addition of information in Bases Section 3/4.4.5 resulted in moving the information for TS Bases Section 3/4.4.6 from page B 3/4 4-3 to page B 3/4 4-4a.

Salem Unit 1 and Unit 2 TS SR 4.2.2.2 contains errors in the use of subscript and superscript for the peaking factor term. It appears that this error was part of the original TS. Standard convention has the subscript followed by the superscript. The subscript and superscript are reversed in some cases in SR 4.2.2.2.

Salem Unit 2 Amendment 197 issued TS 4.2.2.2.c.2 with a typographical error of "CORL" instead of "COLR." "COLR" was used correctly in LCR S94-41 dated May 10, 1996. The original application was supplemented by letters dated March 19, 1997 and August 29, 1997 that provided responses to requests for additional information, but did not change TS 4.2.2.2.c.2. The error appears to have been introduced during the preparation of retyped pages for issuance of Salem Unit 2 Amendment 197. Of note, page 11 of the safety evaluation related to Salem Unit

2 Amendment 197 identified the correction of minor clerical errors to TS 3.1.3.3 and TS 3.1.3.5.

Salem Unit 1 Amendment 84 issued TS Table 3.3-2 title as Reactor Trip System Instrumentation Response Items. The error in the title was introduced in the retyped pages included in LCR S87-05 dated May 5, 1987.

Salem Unit 2 Amendment 244 added a new Limiting Condition for Operation, including an SR, for boron concentration. The SR number was correctly stated in LCR S02-12 dated February 14, 2003. The error in the SR number was introduced in replacement pages included in response to a request for additional information dated October 2, 2003.

Salem Unit 1 Amendment 9 issued TS 6.9.1.5.a including, "Small exposures totalling less than 20% of the individual total dose need not be accounted for." The typographical error was introduced in LCR S77-17 dated October 28, 1977. Salem Unit 1 Amendment 268 eliminated TS 4.4.5.5.b and added new TS 6.9.1.10, "Steam Generator Tube Inspection Report." The failure to delete TS 6.9.1.5.b was an oversight in LCR S04-07 dated February 23, 2005.

4.0 TECHNICAL ANALYSIS

4.1 Elimination of the Requirement for a Power Range Neutron Flux, High Negative Rate Trip

By letter dated May 22, 1987, Westinghouse Owners Group submitted Topical Report WCAP-11394-P, "Methodology for the Analysis of the Dropped Rod Event." In this Topical Report, Westinghouse extended the methodology that was previously used to show that the results of the dropped rod event were acceptable with no credit taken for any direct reactor trip due to dropped rods and with no automatic power reduction features actuated by the dropped rods. Thus, the Power Range, Neutron Flux, High Negative Rate trip could be removed. The NRC reviewed WCAP-11394-P and found the methodology acceptable for referencing in license applications by letter dated October 23, 1989.

In the acceptance of WCAP-11394-P, the NRC stated, "A further review by the staff (for each cycle) is not necessary, given the utility assertion that the analysis described by Westinghouse has been performed and the required comparisons have been made with favorable results." Salem UFSAR Section 15.2.3.2.1 documents the analyses performed in accordance with the methodology of WCAP-11394 for one or more dropped rod cluster control assemblies (RCCAs) and for the dropped RCCA bank. For cases of dropped RCCAs or dropped banks, the DNBR remains greater than the limit value. Therefore, the departure from nucleate boiling (DNB) design criterion is met and the event does not result in core damage. Due to the plant-specific nature of the core physics

characteristics and the thermal-hydraulic dropped rod limit lines, plant-specific data are combined with the appropriate set of correlations and statepoints to verify that the DNB design basis is met for the dropped RCCA event for every Salem fuel cycle design. As committed in Attachment 3 to this submittal, PSEG will use the NRC approved methodology in WCAP-11394-P-A for each fuel cycle to ensure the minimum DNBR is maintained above the DNBR safety limit. Therefore, the proposed changes to Salem TS are consistent with the NRC approved methodology presented in WCAP-11394-P.

Deletion of the requirement for a Power Range, Neutron Flux, High Negative Rate trip would eliminate the potential for an unnecessary reactor trip.

4.2 Correction of TS Errors

The proposed change to Salem Unit 1 Index page XII is editorial in nature, correcting the location of Bases Section 3/4.4.6.

The proposed change to Salem Unit 1 TS 4.2.2.2 is editorial in nature, correcting typographical errors.

The proposed change to Salem Unit 2 TS 4.2.2.2 is editorial in nature, correcting a typographical error and eliminating a condition approved only for Cycle 11, which has expired.

The proposed change to Salem Unit 1 TS Table 3.3-2 is editorial in nature, correcting a typographical error.

The proposed change to Salem Unit 2 SR for boron concentration is editorial in nature, correcting a typographical error.

The proposed change to Salem Unit 1 TS 6.9.1.5.a is editorial in nature, correcting a typographical error. The proposed change to Salem Unit 1 TS 6.9.1.5.b is administrative in nature, correcting information that was superceded by Amendment 268.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed changes discussed in this submittal:

The evaluation focused on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

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

Response: No.

The elimination of the Power Range, Neutron Flux, Negative Rate trip does not increase the probability or consequences of reactor core damage accidents resulting from Rod Cluster Control Assembly (RCCA) Misalignment events previously analyzed. The safety functions of other safety-related systems and components have not been altered. All other Reactor Trip System protection functions are not impacted by the elimination of the requirement for a Power Range, Neutron Flux, High Negative Rate trip. The Power Range, Neutron Flux, High Negative Rate trip circuitry detects and responds to negative reactivity insertion due to RCCA misoperation events, should they occur. Therefore, the Power Range, Neutron Flux, High Negative Rate trip is not assumed in the initiation of such events. The consequences of accidents previously evaluated in the Salem Generating Station (Salem) Updated Final Safety Analysis Report (UFSAR) are unaffected by the proposed changes because no change to any equipment response or accident mitigation scenario has resulted. The proposed changes do not modify the RCCAs or change the acceptance criteria for departure from nucleate boiling (DNB). The TS change reflects analysis described in the UFSAR and cycle-specific analysis performed each fuel cycle.

The proposed revisions to Salem Unit 1 Index page XII, Salem Unit 1 TS 4.2.2.2, Salem Unit 2 TS 4.2.2.2, Salem Unit 1 TS Table 3.3-2, Salem Unit 2 SR number for boron concentration on page 3/4 9-1, Salem Unit 1 TS 6.9.1.5.a, and Salem Unit 1 TS 6.9.1.5.b contain changes administrative in nature that correct errors and do not affect the intent of any TS requirements.

Therefore, the proposed changes do not involve a significant increase in the probability or radiological consequences of an accident previously evaluated.

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

Response: No.

The elimination of the Power Range, Neutron Flux, High Negative Rate trip does not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not challenge the performance or integrity of the RCCAs or any other safety-related system. The proposed changes will have no adverse effect on the availability, operability, or performance of the safety-related systems and components assumed to actuate in the event of a design basis accident (DBA) or transient. It has been demonstrated that the Power Range, Neutron Flux, High Negative Rate trip can be eliminated by the NRC approved methodology described in WCAP-11394-P. The Salem fuel cycle specific analyses have confirmed that for a dropped RCCA event, no direct reactor trip or automatic power reduction is required to meet the DNB limits for this Condition II, "Fault of Moderate Frequency," event. The Power Range, Neutron Flux, High Negative Rate trip is not credited either as a primary or backup mitigation feature for any other UFSAR event.

The proposed revisions to Salem Unit 1 Index page XII, Salem Unit 1 TS 4.2.2.2, Salem Unit 2 TS 4.2.2.2, Salem Unit 1 TS Table 3.3-2, Salem Unit 2 SR number for boron concentration on page 3/4 9-1, Salem Unit 1 TS 6.9.1.5.a, and Salem Unit 1 TS 6.9.1.5.b contain changes administrative in nature that correct errors and do not affect the intent of any TS requirements.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The margin of safety is the difference between the DNB acceptance limit and the failure of the fuel rod cladding. The Salem fuel cycle specific analyses have confirmed that for a dropped RCCA event, DNB limits are not exceeded with the proposed changes. Conformance to the licensing

basis acceptance criteria for DBAs and transients with the elimination of the Power Range, Neutron Flux, High Negative Rate trip is demonstrated and the DNB limits are not exceeded when the NRC approved methodology of WCAP-11394-P is applied. The margin of safety associated with the licensing basis acceptance criteria for any postulated accident is unchanged.

The proposed revisions to Salem Unit 1 Index page XII, Salem Unit 1 TS 4.2.2.2, Salem Unit 2 TS 4.2.2.2, Salem Unit 1 TS Table 3.3-2, Salem Unit 2 SR number for boron concentration on page 3/4 9-1, Salem Unit 1 TS 6.9.1.5.a, and Salem Unit 1 TS 6.9.1.5.b contain changes administrative in nature that correct errors and do not affect the intent of any TS requirements.

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

Based on the above, PSEG concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The following regulatory requirements are applicable:

10 CFR 50.36(c)(2)(ii) – A limiting condition for operation must be included in TS for any item meeting one or more of the following four criteria:

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; and

4. a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

As a result, existing TS requirements that fall within or satisfy any of the criteria in 10 CFR 50.36 must be retained in the TS, while those TS requirements that do not fall within or satisfy these criteria may be relocated to other licensee controlled documents. The Power Range, Neutron Flux, High Negative Rate trip does not meet any of the criteria set out in 10 CFR 50.36 and may therefore be eliminated from TS.

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) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

PSEG has determined the proposed amendment relates to changes in a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or relates to changes in an inspection or a surveillance requirement. However, 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 effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed change is not required.

7.0 REFERENCES

1. Westinghouse Topical Report, WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990

2. Letter from A. C. Thadani (U.S. NRC) to R. A. Newton (Westinghouse Owners Group), "Acceptance for Referencing of Licensing Topical Reports WCAP-11394(P) and WCAP-11395(NP), 'Methodology for the Analysis of the Dropped Rod Event," dated October 23, 1989

3. The NRC has approved a similar license amendment for the Callaway Plant, Unit 1 when it issued Amendment No. 56 on August 23, 1990 (TAC No. 76433)

4. The NRC has approved a similar license amendment for the South Texas Project, Units 1 and 2 when it issued Amendment Nos. 34 and 25 on March 12, 1992 (TAC Nos. M80240 and M80241)

5. Public Service Electric and Gas Company (PSE&G) letter transmitted LCR S77-17 dated October 28, 1977

6. Amendment No. 9 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station Unit No. 1, December 27, 1977

7. PSE&G letter NLR-N87078 transmitted LCR S87-05 dated May 5, 1987

8. Amendment Nos. 84 and 56 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2, November 16, 1987 (TAC Nos. 64857 and 64858)

9. PSE&G letter LR-N96114 transmitted LCR S94-41 dated May 10, 1996

10. PSE&G letter LR-N97040 transmitted responses to Request for Additional Information dated March 19, 1997

11. PSE&G letter LR-N97457 transmitted responses to Request for Additional Information dated August 29, 1997

12. Amendment No. 201 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station Unit No. 1, November 26, 1997

13. Amendment No. 197 to Facility Operating License No. DPR-75 for the Salem Nuclear Generating Station, Unit No. 2, January 8, 1999

14. PSEG Nuclear LLC letter LR-N03-0022 transmitted LCR S02-12 dated February 14, 2003

15. PSEG Nuclear LLC letter LRN-03-0411 transmitted supplement to LCR S02-12 dated October 2, 2003

16. Amendments Nos. 262 and 244 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2, February 6, 2004 (TAC Nos. MC0483 and MC0484)

17. PSEG Nuclear LLC letter LR-N05-0018 transmitted LCR S04-07 dated February 23, 2005

18. Amendment No. 268 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station Unit No. 1, October 14, 2005 (TAC No. MC6213)

19. PSEG Nuclear LLC letter LR-N05-0392 transmitted LCR S05-06 dated August 31, 2005

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

Salem Unit 1 Affected Page List

Index Page XII

The following Technical Specifications for Facility Operating License DPR-70 are affected by this change request:

Technical Specification	Page
Table 2.2-1	2-5
4.2.2.2	3/4 2-6
Table 3.3-1	3/4 3-2
Table 3.3-2	3/4 3-9
Table 4.3-1	3/4 3-11
6.9.1.5.a	6-21
6.9.1.5.b	6-21

The following Technical Specifications for Facility Operating License DPR-75 are affected by this change request:

Technical Specification	Page
Table 2.2-1	2-5
4.2.2.2	3/4 2-6
Table 3.3-1	3/4 3-2
Table 3.3-2	3/4 3-9
Table 4.3-1	3/4 3-11
4.9.2	3/4 9-1

INDEX

BASES

SECTION		PAGE
3/4.3	INSTRUMENTATION	
3/4.3.1	PROTECTIVE	
	AND	
3/4.3.2	ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION	B 3/4 3-1
3/4.3.3	MONITORING INSTRUMENTATION	B 3/4 3-1a
3/4.3.4	TURBINE OVERSPEED PROTECTION	B 3/4 3-4
3/4.4	REACTOR COOLANT SYSTEM	
3/4.4.1	REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	B 3/4 4-1
3/4.4.2	SAFETY VALVES	B 3/4 4-1a
3/4.4.3	RELIEF VALVES	B 3/4 4-1a
3/4.4.4	PRESSURIZER	B 3/4 4-2
3/4.4.5	STEAM GENERATOR (SG) TUBE INTEGRITY	B 3/4 4-2
3/4.4.6	REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-4a~~3~~
3/4.4.7	DELETED	
3/4.4.8	SPECIFIC ACTIVITY	B 3/4 4-5
3/4.4.9	PRESSURE/TEMPERATURE LIMITS	B 3/4 4-6
3/4.4.10	STRUCTURAL INTEGRITY	B 3/4 4-17
3/4.4.11	BLANK	B 3/4 4-17
3/4.4.12	REACTOR VESSEL HEAD VENTS	B 3/4 4-17

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER
	High Setpoint - ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. ~~Power Range, Neutron Flux, High Negative Rate~~ Deleted	~~≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds~~	~~≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds~~
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1865 psig	≥ 1855 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

* Design flow is 82,500 gpm per loop.

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map:

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or
2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the Measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS or the moveable incore detectors when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Comparing the F_{xy} computed (F_{xy}^{C}) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

2. The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP} [1+PF_{xy} (1-P)]$$

Note that this equation was modified by PSEG Submittal LCR S05-06, dated August 31, 2005

where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier for F_{xy} in the COLR, and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional core power distribution measurements shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L}:

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

TABLE 3.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1,2 and *	12
2. Power Range, Neutron Flux	4	2	3	1,2, and 3*	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1,2	2
4. ~~Power Range, Neutron Flux High Negative Rate~~ Deleted	~~4~~	~~2~~	~~3~~	~~1,2~~	~~2~~
5. Intermediate Range, Neutron Flux	2	1	2	1,2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2## and *	4
B. Shutdown	2	0	1	3,4, and 5	5
7. Overtemperature ΔT	4	2	3	1,2	6
8. Overpower ΔT	4	2	3	1,2	6
9. Pressurizer Pressure-Low	4	2	3	1,2	6
10. Pressurizer Pressure--High	4	2	3	1,2	6

TABLE 3.3-2

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~~ Deleted	~~≤ 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 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	R(9)	1, 2, and *
2. Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	Q	1, 2, and 3*
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	Q	1, 2
4. ~~Power Range, Neutron Flux, High Negative Rate~~ Deleted	~~N.A.~~	~~R(6)~~	~~Q~~	~~1, 2~~
5. Intermediate Range, Neutron Flux	S	R(6)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	S(7)	R(6)	Q and S/U(1)	2, 3, 4, 5 and *
7. Overtemperature ΔT	S	R	Q	1, 2
8. Overpower ΔT	S	R	Q	1, 2
9. Pressurizer Pressure--Low	S	R	Q	1, 2
10. Pressurizer Pressure--High	S	R	Q	1, 2
11. Pressurizer Water Level--High	S	R	Q	1, 2
12. Loss of Flow - Single Loop	S	R	Q	1

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation* on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, thermoluminescent dosimeter (TLD), or self reading dosimeter measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. ~~The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b).~~ DELETED

c. The results of any specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555 with a copy to the USNRC Administrator, Region I, no later than the 15th of each month following the calendar month covered by the report.

* This tabulation supplements the requirements of section 20.407 of 10 CFR Part 20.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not applicable	Not applicable
2. Power Range, Neutron Flux	Low setpoint - ≤ 25 % of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER
	High Setpoint - ≤ 109% of RATED THERMAL POWER	High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 second
4. ~~Power Range, Neutron Flux, High Negative Rate~~ Deleted	~~≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second~~	~~≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 second~~
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1865 psig	≥ 1855 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

*Design flow is 82,500 gpm per loop.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map:

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the Measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties~~*~~ as specified in the COLR.

b. Using the PDMS or the moveable incore detectors when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties~~*~~ as specified in the COLR.

c. Comparing the F_{xy} computed (F_{xy}^{C}~~$_{xy}$~~) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}~~$_{xy}$~~) for the appropriate measured core planes given in e. and f., below, and

2. The relationship:

$$F_{xy}^{L}\text{~~}_{xy}\text{~~} = F_{xy}^{RTP}\text{~~}_{xy}\text{~~} \quad [1 + PF_{xy}(1-P)]$$

where F_{xy}^{L}~~$_{xy}$~~ is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}~~$_{xy}$~~, PF_{xy} is the power factor multiplier for F_{xy} in the COLR~~L~~, and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^{C}~~$_{xy}$~~ is greater than the F_{xy}^{RTP}~~$_{xy}$~~ limit for the appropriate measured core plane but less than the F_{xy}^{L}~~$_{xy}$~~ relationship, additional core power distribution measurements shall be taken and F_{xy}^{C}~~$_{xy}$~~ compared to F_{xy}^{RTP}~~$_{xy}$~~ and F_{xy}^{L}~~$_{xy}$~~:

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C}~~$_{xy}$~~ was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

~~*For Cycle 11, when the number of available movable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/14.5)(1%)] where T is the number of available thimbles.~~

TABLE 3.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1,2 and *	12
2. Power Range, Neutron Flux	4	2	3	1,2 and 3*	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1,2	2
4. ~~Power Range, Neutron Flux, High Negative Rate~~ Deleted	~~4~~	~~2~~	~~3~~	~~1,2~~	~~2~~
5. Intermediate Range, Neutron Flux	2	1	2	1,2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##, and *	4
B. Shutdown	2	0	1	3,4 and 5	5
7. Overtemperature ΔT	4	2	3	1,2	6
8. Overpower ΔT	4	2	3	1,2	6
9. Pressurizer Pressure-Low	4	2	3	1,2	6
10. Pressurizer Pressure--High	4	2	3	1,2	6

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~~ Deleted	~~≤ 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 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	R[9]	1, 2, and *
2. Power Range, Neutron Flux	S	D[2], M[3] and Q[6]	Q	1, 2, and 3*
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R[6]	Q	1, 2
4. ~~Power Range, Neutron Flux, High Negative Rate~~ Deleted	~~N.A.~~	~~R[6]~~	~~Q~~	~~1, 2~~
5. Intermediate Range, Neutron Flux	S	R[6]	S/U[1]	1, 2 and *
6. Source Range, Neutron Flux	S[7]	R[6]	Q and S/U[1]	2, 3, 4, 5 and *
7. Overtemperature ΔT	S	R	Q	1, 2
8. Overpower ΔT	S	R	Q	1, 2
9. Pressurizer Pressure--Low	S	R	Q	1, 2
10. Pressurizer Pressure--High	S	R	Q	1, 2
11. Pressurizer Water Level--High	S	R	Q	1, 2
12. Loss of Flow - Single Loop	S	R	Q	1

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of the Reactor Coolant System, the fuel storage pool, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 6 (Only applicable to the refueling canal, the fuel storage pool and refueling cavity when connected to the Reactor Coolant System)

ACTION:
With the requirements of the above specification not satisfied, immediately

a. Suspend CORE ALTERATIONS and
b. Suspend positive reactivity additions and
c. Initiate action to restore boron concentration to within limit specified in the COLR.
d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1~~2~~ Verify the boron concentration is within the limit of the COLR every 72 hours.

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by PSEG Nuclear LLC (PSEG) in this document. Any other statements in this submittal are provided for information only purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. Paul Duke at (856) 339-1466.

Regulatory Commitment	Due Date/Event
PSEG will use the NRC approved methodology in WCAP-11394-P-A for each fuel cycle to ensure the minimum DNBR is maintained above the DNBR safety limit.	Concurrent with implementation of the amendment

PROPOSED CHANGES TO TS BASES PAGES

The following Technical Specifications Bases for Salem Generating Station Unit 1 and Unit 2, Facility Operating License Nos. DPR-70 and DPR-75 are affected by this change request:

Technical Specification	Page
Bases 2.2.1	B 2-4

automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rate~~s~~

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

~~The Power-Range Neutron Flux Negative Rate trip channels were originally designed to provide trip signals when needed to ensure that the minimum DNBR was maintained above the safety limit for all single or multiple rod drop events. No credit is taken for operation of this trip in the accident analysis; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System. Current safety analysis shows that the minimum DNBR never falls below the safety limit in any RCCA drop event.~~

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Δ T

The Overtemperature Δ T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

matically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rate~~s~~

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

~~The Power Range Neutron Flux Negative Rate trip channels were originally designed to provide trip signals when needed to ensure that the minimum DNBR was maintained above the safety limit for all single or multiple rod drop events. No credit is taken for operation of this trip in the accident analysis; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System. Current safety analysis shows that the minimum DNBR never falls below the safety limit in any RCCA drop event.~~

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ~~Delta~~ Δ T

The Overtemperature ~~delta~~ Δ T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.