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MAR 1 6 2007

10 CFR 50.90

LR-N07-0030

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

SALEM GENERATING STATION – UNIT 1 AND UNIT 2
FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75
NRC DOCKET NOS. 50-272 AND 50-311

Subject: LICENSE CHANGE REQUEST S06-010, STEAM GENERATOR FEEDWATER PUMP TRIP, FEEDWATER ISOLATION VALVE RESPONSE TIME TESTING and CONTAINMENT COOLING SYSTEM

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

This License Change Request (LCR) involves (1) new TS surveillance requirements for Steam Generator Feedwater Pump (SGFP) trip and Feedwater Isolation Valve (FIV) closure, and (2) revised TS surveillance requirements for Containment Fan Cooler Unit (CFCU) flow. The LCR relates to adoption of a new containment response analysis that credits Steam Generator Feedwater Pump (SGFP) Trip and Feedwater Isolation Valve closure (on a feedwater regulator valve failure) to reduce the mass/energy release to containment during a Main Steam Line Break (MSLB). The containment analysis also credits a reduced heat removal capability for the Containment Fan Cooler Units (CFCUs), allowing a reduction in the required Service Water (SW) flow to the CFCUs.

The containment pressure/temperature response was re-analyzed in WCAP-16503, Revision 3, "Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit Margin Recovery Project," February 2007." The Containment Integrity Analyses were performed for design-basis LOCA and MSLB transients to determine the acceptability of reducing the accident CFCU heat removal capacity. The proposed change affects the post-accident

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operation of the containment heat removal systems. The methods used for the analyses incorporate modeling refinements consistent with current licensed methodology for LOCA and MSLB containment response.

The requested changes would support planned simplification of the Service Water System design to the CFCUs for Salem Units 1 and 2 to improve system reliability, and would also be applicable to Salem Unit 2 following the steam generator replacement planned for refueling outage 2R16 in Spring, 2008.

This submittal impacts TS Table 3.3-5 (TS 3.3.2.1) and TS Surveillance 4.6.2.3.b.3 and 4.6.2.3.c.2:

TS 3.3.2.1 states: "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 [Engineered Safety Features Response Times]." The response times for the SGFP and the FIV will be added to Table 3.3-5; currently the response time for feedwater isolation only includes the response time (10 seconds) for the feedwater regulator valve (FRV).

TS Surveillances 4.6.2.3.b.3 and 4.6.2.3.c.2 establish the requirements for verifying CFCU cooling water flow rates. The flow rates will be changed to reflect the new containment analyses.

Attachment 1 provides a description of the proposed changes. Attachment 2 provides the existing TS pages marked up to show the proposed changes. For information, Attachment 3 provides the existing TS Bases pages marked-up to reflect the associated changes to the TS Bases. Enclosure 1 provides WCAP-16503-NP, Revision 3. Enclosure 2 provides the Engineering Evaluation (S-C-CBV-MEE-1982, Revision 0) justifying the acceptability of WCAP-16503, Revision 3 for use in support of the proposed Technical Specification changes.

PSEG requests the license amendment by March 1, 2008 to support refueling outage 2R16 with an implementation period tied to outage 2R16 (1R19 for Unit 1).

If you have any questions or require additional information, please do not hesitate to contact Mr. Jamie Mallon at (610) 765-5507.

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

Executed on 3/16/07
(Date)

Sincerely,

Thomas P. Joyce
Site Vice President
Salem Generating Station

Attachments (3)
Enclosures (2)

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS STEAM GENERATOR FEEDWATER PUMP TRIP, FEEDWATER ISOLATION VALVE RESPONSE TIME TESTING and CONTAINMENT COOLING SYSTEM

Table of Contents

CHANGES TO TECHNICAL SPECIFICATIONS

1. DESCRIPTION

The purpose of this License Change Request (LCR) is to (1) add the Steam Generator Feedwater Pump (SGFP) trip and Feedwater Isolation Valve (FIV) RESPONSE TIMES to the TS, and (2) revise the Containment Fan Cooler Unit (CFCU) cooling water flow rate TS requirement. The LCR relates to adoption of a new containment response analysis that credits Steam Generator Feedwater Pump (SGFP) Trip and Feedwater Isolation Valve closure (on a feedwater regulator valve failure) to reduce the mass/energy release to containment during a Main Steam Line Break (MSLB). The containment analysis also credits a reduced heat removal capability for the Containment Fan Cooler Units (CFCUs), allowing a reduction in the required Service Water (SW) flow to the CFCUs.

The containment pressure/temperature response was re-analyzed in WCAP-16503, Revision 3, "Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit Margin Recovery Project," February 2007." The Containment Integrity Analyses were performed for design-basis LOCA and MSLB transients to determine the acceptability of reducing the accident CFCU heat removal capacity. The proposed change affects the post-accident operation of the containment heat removal systems. The methods used for the analyses incorporate modeling refinements consistent with current licensed methodology for LOCA and MSLB containment response.

The requested changes would support planned simplification of the Service Water System design to the CFCUs for Salem Units 1 and 2 to improve system reliability, and would also be applicable to Salem Unit 2 following the steam generator replacement planned for refueling outage 2R16 in Spring, 2008.

Currently, TS for feedwater isolation only include the response time for the feedwater regulator valves (FRV), which is 10 seconds. The proposed change will include:

- Closure of the FRVs after 10-seconds, and the FIVs after 32-seconds.
- Trip of SGFPs after a 7-second signal processing and mechanical delay, not including pump coastdown time.

Currently, TS for CFCU operability requires a minimum flow rate of 2550 gpm. The proposed change will reduce the minimum flow rate to 1300 gpm.

2. PROPOSED CHANGE

This submittal proposes changes to TS Table 3.3-5 (TS 3.3.2.1) and TS Surveillance 4.6.2.3.b.3 and 4.6.2.3.c.2:

"TS 3.3.2.1 states:

"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 [Engineered Safety Features Response Times]."

The response times (for the initiating signals that are required by the containment response analysis[1]) for the SGFP trip and the FIV closure will be added to Table 3.3-5; currently the response time for feedwater isolation, for all Initiating Signals, only includes the response time for the feedwater regulator valve (FRV) and FRV bypass valve closure. This change will be accomplished by replacing the current values for feedwater isolation response time in Table 3.3-5 with the following note (Note 8):

"Feedwater isolation includes closure of the feedwater regulating valves (FRV), the FRV bypass valves, the feedwater isolation valves (FIV), and trip of the steam generator feedwater pumps (SGFP). The response time for feedwater isolation by closure of the FRVs (the BF-19 valves) and the FRV bypass valves (the BF-40 valves) is 10 seconds. The response time for feedwater isolation by closure of the FIVs (the BF-13 valves) is 32 seconds. The response time for feedwater isolation by trip of the SGFPs is 7 seconds, not including pump coastdown time."

As discussed above, the current response time for feedwater isolation, for all Initiating Signals, includes the response time for the feedwater regulator valve (FRV) and FRV bypass valve closure. An additional note (Note 9) will be added to Table 3.3-5 for the Initiating Signals for feedwater isolation that are not impacted by the proposed change, clarifying that their Response Time only credits closure of the FRVs (BF-19 valves) and FRV bypass valves (BF-40 valves).

TS 4.6.2.3, Surveillance 4.6.2.3.b.3 states:

[At least once per 31 days by:] Verifying a cooling water flow rate of greater than or equal to 2550 gpm to each cooler.

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[1] Containment Pressure - High, Differential Pressure between Steam Lines - High, Steam Flow high coincident with Steam Pressure Low

TS 4.6.2.3.b.3 will be changed to read: *"Verifying a cooling water flow rate of greater than or equal to 1300 gpm to each cooler."*

TS 4.6.2.3, Surveillance 4.6.2.3.c.2 states:

[At least once per 18 months by verifying that on a safety injection test signal:] The automatic valves and dampers actuate to their correct positions and that the cooling water flow rate to each cooler is greater than or equal to 2550 gpm.

TS 4.6.2.3.c.2 will be changed to read: *"The automatic valves and dampers actuate to their correct positions and that the cooling water flow rate to each cooler is greater than or equal to 1300 gpm."*

3. BACKGROUND

Elements of the existing containment analyses for Salem Unit 1 and 2, and the existing design of the CFCU flowpath are discussed below

a. Current containment analysis inputs and assumptions, regarding the SGFP trip and FIV closure, are described below.

Following a Main Steam Line Break (MSLB), feedwater (FW) flow is isolated to limit the mass and energy released into the containment through the failed steam line. FW isolation includes the following: closure of the FRVs and FRV bypass control valves (BF-19s and BF-40s); trip of the SGFPs due to closure of the High Pressure Turbine Stop Valves (MS43) or Low Pressure Turbine Stop Valve (RS15); and closure of the motor-operated FIVs (BF-13s). UFSAR Figure 10.4-5B Sheet 3 depicts the FRV and FIV arrangement and UFSAR Figure 10.3-1B Sheet 3 shows the SGFP turbine stop valves.

As described in UFSAR Section 15.4.8.2.2, the FRVs close within 10 seconds of any Safety Injection (SI) signal. The FIVs close within 32 seconds of any SI signal. Therefore, FRV failure results in an additional 22 seconds during which feed water from the Condensate/Feed System may be added to the faulted steam generator. Also, since the feed water isolation valve is upstream of the regulator valve, failure of the regulator valve results in additional feedline volume that is not isolated from the faulted steam generator. Thus, water in this portion of the lines can flash and enter the faulted steam generator.

Feed water flow to the faulted steam generator from the Main Feed Water System is calculated using the hydraulic resistances of the system piping, head/flow curves for the main feed water pumps, and the steam generator pressure decay as calculated by the LOFTRAN code. Assumptions used to conservatively maximize the calculated flows include:

- No credit is taken for extra pressure drop in the feedlines due to flashing of water.
- Feed water regulator valves in the intact loops do not change position prior to a trip signal.
- All feed water pumps are running at maximum speed.

While the approved MSLB accident methodology allows crediting closure of the BF-13 FIV to terminate flow and trip of the SGFP to reduce flow until the BF-13 FIV is fully shut, the current containment analyses do not credit the reduced mass and energy release that would occur with the SGFP trip and while the BF-13 FIVs close. This is conservative because the analyses assume full FW flow to the faulted SG for 32 seconds, until the BF-13 MOV was fully shut.

b. Current CFCU design is discussed below.

The existing CFCU flow path design is very complex. The dual mode flow control contains three active air operated control valves in each the five CFCU flow paths. Each of these valves must change state on an accident signal and the outlet control valve, SW-223, must switch between two different flow control setpoints (normal/low to accident/high flow).

PSEG intends to use the revised containment analyses, with the reduced CFCU flows, to support simplification of the CFCU / SW design.

4. TECHNICAL ANALYSIS

<u>WCAP-16503</u>

WCAP-16503, Revision 3, addresses the long term Loss of Coolant Accident (LOCA) and worst case MSLB scenarios. Previous efforts to investigate margin recovery options concluded that the analysis of MSLB events with an appropriate spectrum of cases (break size, limiting single-failure scenario, and power level) was needed. Also, the duration of the analyses for the new LOCA cases were extended to 1×10^7 seconds (approximately 120 days) to support the Environmental Qualification bases for critical equipment. The new analyses also account for the Salem Unit 2 steam generator replacement project, scheduled to be implemented in Spring 2008.

The LOCA and MSLB cases for Salem Unit 1 and Unit 2 are analyzed with the current licensing-basis methods and analysis tools that have been reviewed and approved for the Salem units several times over the duration of plant operation. The Westinghouse steamline break mass and energy release methodology was approved by the NRC (Reference 8 of WCAP) and is documented in WCAP-8822, "Mass and Energy Releases Following a Steam Line Rupture" (Reference 9 of WCAP). WCAP-8822 forms the basis for the assumptions and models used

in the calculation of the mass and energy releases resulting from a steamline rupture.

Upon a break, WCAP-16503 assumes pertinent steamline break protection systems actuate a safety injection (SI) signal that starts the SI pumps and will also result in:

- Reactor trip after 2-seconds.
- Start of auxiliary feedwater (no delay).
- Closure of the faulted loop FRV after 10-seconds,
- Closure of the FIVs after 32-seconds.
- Trip of SGFPs after a 7-second signal processing and mechanical delay, with a 7-second coastdown.
- Start of CFCUs after 100-seconds.

The FRV on each of the intact SGs is conservatively assumed to close at the time of the SI signal which terminates the main FW flow to the intact SGs. However, the FRV on the faulted loop is assumed to open quickly in response to the steamline break before getting a signal to close. Starting at 0.2 second, the main feedwater flowrate modeling is based on the faulted loop FRV fully open (and the intact loop FRVs fully closed). For normal response (with no assumed component failures), FW is added to the faulted SG until the faulted loop FRV closes, which occurs 10 seconds after the SI signal.

For the single failure scenario of the faulted loop FRV failing open, the WCAP now assumes that the SGFPs are tripped after 7-seconds (instrument and mechanical delays) and have a 7-second coastdown. However, because the condensate pumps are not tripped from an SI signal, pumped flow (from the condensate pumps through the SGFPs) continues until the FIV is fully closed 32 seconds after the SI signal. In essence, the WCAP revises the amount of flow from the SGFPs through the faulted loop for the period up to closure of the FIV.

The revised modeling remains in conformance with the UFSAR. As indicated above, UFSAR Section 15.4.8.2 states "the worst failure in this system is a failure of the main feed water regulator valves to close. This failure results in an additional 22 seconds during which feed water from the Condensate Feed System may be added to the faulted steam generator."[2] The 22 seconds is the

[2] In the current Analysis of Record (AOR), the FRV failure was limiting for pressure in the overall MSLB analysis (UFSAR page 15.4-121). However, per WCAP 16503 Revision 3, following Unit 2 SG replacement, the overall limiting MSLB failure case for containment pressure has changed from the FRV to a Containment Safeguards Failure (CSF).

difference between the full closure of the FRV (10 seconds) and full closure of the FIV (32 seconds). Although the UFSAR covers this modeling sequence, the original accident modeling was overly conservative as it assumed full FW flow to the faulted SG for 32 seconds, until the FIV was fully shut. The revised modeling with WCAP-16503 credits reduced flow when the SGFP is tripped. It further reduces the flow as FIV closure increasingly throttles the flow from the Condensate Pumps.

During the SGFP coast-down, FW flow is assumed to decrease linearly to the flowrate provided by the condensate pumps (through freewheeling SGFPs). The baseline SGFP and condensate pump flowrates are calculated using Westinghouse's LOFTRAN analysis program and are a function of the transient pressure in the faulted SG. Recent plant data from a Salem Unit 2 manual reactor trip from approximately 94% power (9/26/06 event described in Licensee Event Report LER 311/06-003 dated November 27, 2006) shows that the SGFP discharge header flow decreases linearly and is interrupted in approximately 5-6 seconds due to SGFP discharge check valves closing from reverse feedwater pressure differentials. The SGFP discharge check valves reopen several seconds later as feedwater discharge header pressure decreases below the operating condensate header pressure and significantly lower feedwater flow is reintroduced to the steam generators.

Closure of FIV terminates the remaining pump flow from the condensate system. This is modeled assuming a 2-second electronic delay before initiation of the valve closure, 20 seconds of valve closure that have no impact on the FW flowrate, and a linear flowrate reduction during the final 10 seconds of the valve stroke. This design basis stroke time for the FIVs of 30 seconds is conservative. Results from past tests indicate actual stroke times of about 26 seconds. In addition, testing per Surveillance Procedure S1(2).OP-ST.MS-0002 verifies actuation and closure of the BF13 valves occur in less than or equal to 29.0 seconds.

Lower CFCU heat removal rates (as a function of containment temperature) are assumed in the WCAP to allow future planned modifications or analysis revisions such as increased plugging, justify increased fouling factor, and/or reduced SW accident flow rates. There is no reduction in the number of CFCUs which will be retained operable. For those cases where the single failure is the loss of a safeguards train, three CFCUs and one containment spray pump are available for containment cooling. For other single-failure scenarios, the maximum number of CFCUs could be five and the maximum number of spray pumps could be two.

WCAP-16503 Inputs and Assumptions

The containment analysis inputs and assumptions have been changed in WCAP-16503 for the SGFPs and CFCUs relative to previous containment analyses. This included revision to the analyses for the single failure case when the FRV to the faulted SG is assumed to fail open, and a reduction in the assumed CFCU heat removal capacity. The relevant input plant parameters for this change include:

- SI signal trip of the FIV's and any associated time delays.
- SI signal trip of the SGFPs and any associated time delays.
- SGFP coastdown time and flow/pressure characteristics.
- Flow characteristics during FIV closure.
- Minimum CFCU heat removal capacity.

The new inputs/assumptions and corresponding bounding plant parameters have been justified and documented in Enclosure 2 (S-C-CBV-MEE-1982, Revision 0, "Updated Containment Pressure / Temperature Response Analysis With SGFP Trip). The SGFP coast down time assumption discussed in Enclosure 2 has been further evaluated. An analysis of the plant trip on September 26, 2006 indicates that the SGFP coast down time is approximately 5-6 seconds after the pumps are tripped, therefore the WCAP assumption of a 7 second coast down time is appropriate.

Additional analyses refinements are discussed in Section 6.2 of Enclosure 2.

WCAP Containment Temperature/Pressure Responses

A comparison of the revised containment temperature/pressure responses calculated in WCAP-16503 to the current analysis of record (AOR) are shown in the table below. Analysis results for Salem Unit 2 reflect the AREVA Model 61/19T Replacement Steam Generators (RSGs) planned for installation during the Spring 2008 refueling outage.

Containment Analysis Peak Conditions

DBA Case	AOR Peak Press.	Revised Peak Press. WCAP-16503	AOR Peak Temp.	Revised Peak Temp. WCAP-16503
LOCA	41.2 psig	U1: 40.9 psig U2: 43.5 psig	263.3 °F	U1: 262.1 °F U2: 265.9 °F
MSLB	45.1 psig	U1: 41.0 psig U2: 45.6 psig	351.0 °F	U1: 348.2°F U2: 349.6°F

The revised peak pressure and temperature values remain below the maximum values of 47 psig and 351.3 °F specified in TS 5.2.2.

Impact of WCAP-16503 Temperature / Pressure Response

The impact of the new analyses on plant systems and components has been evaluated to ensure design capacities provide sufficient margin in bounding the containment analysis results. The following areas have been addressed.

- Overall Containment Response
- Impact on UFSAR Chapter 15 Events and Other ECCS/Heat Removal Systems
- Impact on Dose Analyses
- Impact on Appendix J Test Requirements
- Impact on Environmental Qualification (EQ) of Equipment Inside Containment
- Impact on CFCU SW Piping Temperature Analyses
- Impact on Pressure Locking/Thermal Binding (PL/TB) Analyses
- Impact on Generic Letter 96-06 Analyses

Enclosure 2 documents that the WCAP-16503 containment temperature/pressure responses do not adversely impact the design basis, the licensing basis or related analyses, and support the proposed changes to the Technical Specifications.

5. REGULATORY SAFETY ANALYSIS

The purpose of this License Change Request (LCR) is to (1) add the Steam Generator Feedwater Pump (SGFP) trip and Feedwater Isolation Valve (FIV) RESPONSE TIMES to the TS, and (2) reduce the Containment Fan Cooler Unit (CFCU) cooling water flow rate TS requirement, both changes resulting from a new containment analysis.

5.1 No Significant Hazards Consideration

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

Response: No

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change establishes response time requirements for feedwater isolation and reduced CFCU flow rates to support containment analyses to accommodate reduced CFCU heat removal capacity. The changes in analysis input assumptions affect plant response to an accident and are not accident initiators; therefore, they have no bearing on the probability of an accident. The Salem FSAR Chapter 15 accidents which are impacted by a change in the CFCU modeling parameters are LOCA and MSLB mass and energy release Containment analyses. The consequences of these postulated accidents are shown to be acceptable using assumptions consistent with the proposed changes.

For the LOCA transients, the containment cooling systems are considered for three aspects: core response, containment response and dose. The core response is most limiting when the containment conditions minimize back pressure since this increases the blowdown and reduces the effectiveness of the ECCS. The LOCA core response (10CFR50.46 – PCT) is conservatively biased to minimize the containment backpressure such that any safety injection effectiveness is minimized (the core becomes the highest resistance flow path). Thus, any reduction in the accident capability of the CFCUs has no bearing on the LOCA core response.

The bounding containment integrity analyses are the LBLOCA and the MSLB Inside Containment events. The containment integrity analysis relies on two heat removal paths to maintain containment pressure and temperature conditions. The CFCU air-to-water heat exchangers reject containment energy to the SW System and the Containment Spray System removes containment energy by using spray droplet direct contact

heat exchange to transfer the energy from the containment ambient to the containment sump, where it is transferred out of containment via the RHR heat exchanger and CCW/SW Systems. Containment integrity analyses for both LOCA and MSLB, using input assumptions consistent with the proposed changes, show that containment integrity is maintained with reduced CFCU heat removal capacity.

The potential dose impacts due to reduced CFCU heat removal capacity are bounded as the design basis assumptions concerning the number of operating CFCUs (three of five), and the thermal-hydraulic transient operation of the Containment Spray System are unchanged. The Salem design basis only credits Containment Spray iodine removal effectiveness during the LOCA injection and recirculation phases based on a single failure of an entire ESF train. This assumption results in 3 of 5 CFCUs being available to ensure adequate mixing of the containment ambient air as well as operation of a single Containment Spray Train, which controls containment spray droplet size and pH, as described in UFSAR Section 6.2.3. As a further conservatism, the current LOCA Alternate Source Term (AST) analysis (Calculation S-C-ZZ-MDC-1945, an interim revision of which was sent to the NRC staff for review via letter dated September 16, 2004) only credits two CFCUs for mixing. The Containment Building and Auxiliary Building leakage rates are unaffected by the revised containment analysis as the peak containment pressure and temperatures are less than the design basis values described in the Salem UFSAR. Therefore, there is no impact on offsite dose rates due to the reduced CFCU heat removal capacity.

One other high energy line break for consideration is the rupture of a feedwater line break. From a containment response aspect, this event is bounded by the MSLB event, so it is not explicitly analyzed (or even discussed in the Salem UFSAR).

A review of the Salem design basis for AST dose calculations shows that the revised Containment Integrity Analysis, WCAP-16503, does not challenge any of the assumptions that are part of the AST design basis.

Section 6.2 of the UFSAR indicates that the Appendix J Type A containment leak rate test pressure is based on the containment design pressure of 47.0 psig, not the calculated accident pressure. Since the design pressure value bounds the peak pressure calculated in WCAP-16503 and is not being changed, the Appendix J testing requirements are not impacted.

Thus, in conclusion, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change modifies response time requirements

for feedwater isolation, and reduces CFCU flow rates and heat removal requirements consistent with the new containment analysis.

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

Response: No

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes support revised containment analysis to accommodate the reduced CFCU heat removal capacity.

The response time-related changes impose new surveillance acceptance criteria to existing plant equipment that actuates to isolate feedwater following a safety injection signal. There is no change in actuation logic associated with the addition of response time criteria; therefore no new accident sequences would result from the imposition of response time test criteria to existing plant equipment.

The reduction in minimum service water system flow to the CFCUs is supported by analyses demonstrating acceptable system performance and containment integrity following a demand for system operation. The post-accident conditions resulting from the proposed reduction in flow do not adversely impact the environmental qualification of equipment, such that no new consequential failures are introduced to any design basis accident scenario. CFCU operation with the proposed reduction in minimum required accident flow would not result in the progression of any design basis event into a previously unanalyzed accident. Therefore, no new accident scenarios are created from the CFCU flowrate reduction.

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

Response: No

The proposed change does not involve a significant reduction in the margin of safety. The revised containment analyses accommodate reduced CFCU heat removal capacity using input assumptions consistent with the proposed changes.

The proposed change involves the addition of feedwater isolation response time surveillance criteria and reduction in minimum service water system flows to CFCUs. These changes affect input to the analyses of mass/energy releases and containment response to a design basis main steam line break or loss of coolant accident. The analyses, consistent with

the proposed changes, demonstrate that the acceptance criteria continue to be met, and the post-accident conditions do not adversely affect containment integrity or otherwise challenge any safety limit. The margin of safety with respect to containment pressure is preserved by demonstrating that the calculated pressures do not exceed the design limit of 47 psig.

Based on the above, PSEG concludes that the proposed change presents no significant hazards 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

Note: The general design criteria that were followed in the design of Salem Unit 1 and 2 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. Salem Unit 1 and 2 design also conforms to the intent of 10CFR50, Appendix A ("General Design Criteria for Nuclear Power Plants," dated July 7, 1971). The applicable 10 CFR 50, Appendix A General Design Criterion are listed below.

Criterion 16--Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 50--Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 38--Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

10 CFR 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

Appendix J to Part 50--Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Option B--Performance-Based Requirements

One of the conditions required of all operating licenses for light-water-cooled power reactors as specified in § 50.54(o) is that primary reactor containments meet the leakage-rate test requirements in either Option A or B of this appendix. These test requirements ensure that (a) leakage through these containments or systems and components penetrating these containments does not exceed allowable leakage rates specified in the Technical Specifications and (b) integrity of the containment structure is maintained during its service life. Option B of this appendix identifies the performance-based requirements and criteria for preoperational and subsequent periodic leakage-rate testing.

Meeting the above requirements, as documented in WCAP-16503, Revision 3 (Enclosure 1) and S-C-CBV-MEE-1982, Revision 0 (Enclosure 2), ensure all containment design limits remain bounded under the CFCU margin recovery program.

GL 89-13 and GL 96-06 Commitments

The change in the required Service Water (SW) flow for the CFCUs will not impact any commitments with respect to Generic Letters 89-13 and 96-06.

NRC Generic Letter 89-13:

Based on operating experience at facilities using open cycle service water systems, NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated July 18, 1989, provides programmatic recommendations to meet General Design Criteria (GDC) 44—"Cooling Water" that requires provision of a system "to transfer heat from structures, systems, and components important to safety to an ultimate heat sink" (UHS): "GDC 45--Inspection of Cooling Water System" that requires the system design "to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system;" and "GDC 46--Testing of Cooling Water System" that requires the design "to permit appropriate periodic pressure and functional testing."

Salem Generating Station CFCUs use an open cycle service water system and are subject to Generic Letter 89-13. Salem GL 89-13 commitments include ensuring heat exchangers cooled by the SW System are capable of performing their design function and performing inspections of system components. Periodic preventative maintenance (PM) performed on the CFCUs with respect to these commitments includes: (1) open, inspect and cleaning of the tubes, tubesheets and waterboxes; (2) bioufouling montoring (differential pressure monitored as a function of flow); (3) high flow flushing. All of these PMs will continue to be performed. No changes to the current frequencies of these PMs are planned; any potential changes would occur post-implementation based on trending in accordance with the GL 89-13 Program.

NRC Generic Letter 96-06:

NRC Generic Letter (GL) 96-06, "Assurance Of Equipment Operability And Containment Integrity During Design-Basis Accident Conditions" dated September 30, 1996 and GL 96-06 Supplement 1 dated November 13, 1997, address the following concerns:

"1. Cooling water systems serving the containment air coolers may be exposed to the hydrodynamic effects of waterhammer during either a loss-of-coolant accident (LOCA) or a main steam line break (MSLB). These cooling water systems were not designed to withstand the hydrodynamic effects of waterhammer and actions may be needed to satisfy system design and operability requirements.

2. Cooling water systems serving the containment air coolers may experience two-phase flow conditions during postulated LOCA and MSLB scenarios. The heat removal assumptions for design-basis accident scenarios are based on single-phase flow conditions and actions may be needed to satisfy system design and operability requirements.

3. Thermally induced overpressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could lead to a breach of containment integrity through bypass leakage. Actions may be needed to satisfy system operability requirements."

Salem Generating Station GL 96-06 commitments include: (1) ensuring no waterhammer or two-phase flow in the SW System during accident conditions; and (2) evaluation of piping penetrations for potential thermally induced overpressurization during normal and accident conditions. The proposed new CFCU flow requirement of 1300 gpm was based on ensuring no two-phase flow in the CFCU SW discharge piping, as well as meeting the required CFCU heat removal requirement, as evaluated in Enclosure 2. The SW Accumulator vessels

will remain in place. Enclosure 2 also evaluated the impact on potential overpressurization of piping penetrations based on the revised containment analysis from Enclosure 1, and concluded that penetrations either have overpressure protection or the pressure increases remain below the maximum allowable pressures.

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 Commissions' regulations; and
3) Issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

6. ENVIROMENTAL CONSIDERATIONS

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. 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 off site, or (iii) a significant increase in individual or cumulative occupational 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. REFERENCES

7.1 WCAP-16503, Revision 3, "Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit Margin Recovery Project", February 2007.
7.2 S-C-CBV-MEE-1982, Revision 0, "Updated Containment Pressure / Temperature Response Analysis With SGFP Trip
7.3 S-C-ZZ-MDC-1945, Revision 0, "Post LOCA EAB, LPZ and CR Doses – Alternative Source Term (AST)

SALEM GENERATING STATION UNIT 1 and UNIT 2
FACILITY OPERATING LICENSE NO. DPR-70 and NO. DPR-75
DOCKET NO. 50-272 and NO. 50-311
REVISIONS TO THE TECHNICAL SPECIFICATIONS

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License DPR-70 (Salem Unit 1) are affected by this change request:

Technical Specification	Page
Table 3.3-5 (TS 3.3.2.1)	3/4 3-27 through 31
TS 4.6.2.3	3/4 6-11 and 11a

The following Technical Specifications for Facility Operating License DPR-75 (Salem Unit 2) are affected by this change request:

Technical Specification	Page
Table 3.3-5 (TS 3.3.2.1)	3/4 3-28 through 32
TS 4.6.2.3	3/4 6-12 and 13

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)	≤27.0(1)
b. Reactor Trip (from SI)	≤2.0
c. Feedwater Isolation	~~≤10.0~~ Note 8
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 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 Note ~~8~~9
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~~Note 8
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 Note ~~8~~9
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 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~~Note 8
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 Note ~~8~~9
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

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

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.

(8) Feedwater isolation includes closure of the feedwater regulating valves (FRV), the FRV bypass valves, the feedwater isolation valves (FIV), and trip of the steam generator feedwater pumps (SGFP). The response time for feedwater isolation by closure of the FRVs (the BF-19 valves) and the FRV bypass valves (the BF-40 valves) is 10 seconds. The response time for feedwater isolation by closure of the FIVs (the BF-13 valves) is 32 seconds. The response time for feedwater isolation by trip of the SGFPs is 7 seconds, not including pump coastdown time.

(9) Feedwater isolation includes closure of the feedwater regulating valves (FRV) and the FRV bypass valves.

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Five containment cooling fans shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or two of the above required containment cooling fans inoperable, restore the inoperable cooling fan(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With three or more of the above required containment cooling fans inoperable, restore at least three cooling fans to OPERABLE status within 1 hour or be in at least HOT STANDBY WITHIN the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the remaining inoperable cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment cooling fan shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 12 hours by:

1. Verifying the water level in each service water accumulator vessel is greater than or equal to 226 inches and less than or equal to 252 inches.

2. Verifying the temperature in each service water accumulator vessel is greater than or equal to 55°F and less than or equal to 95°F.

3. Verifying the nitrogen cover pressure in each service water accumulator vessel is greater than or equal to 135 psig and less than or equal to 160 psig.

b. At least once per 31 days by:

1. Starting (unless already operating) each fan from the control room in low speed.

2. Verifying that each fan operates for at least 15 minutes in low speed.

3. Verifying a cooling water flow rate of greater than or equal to ~~2550~~ 1300 gpm to each cooler.

c. At least once per 18 months by verifying that on a safety injection test signal:

1. Each fan starts automatically in low speed.

2. The automatic valves and dampers actuate to their correct positions and that the cooling water flow rate to each cooler is greater than or equal to ~~2550~~ 1300 gpm.

d. At least once per 18 months by verifying that on a loss of offsite power test signal, each service water accumulator vessel discharge valve response time is within limits.

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)	≤ 27.0[1]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	~~≤ 10.0~~ Note 8
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 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 Note 9~~8~~
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~~ Note 8
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 Note 9~~8~~
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 3.3-5 (Continued)

ENGINEERE 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~~ Note 8
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 Note 9~~8~~
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

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

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.

(8) Feedwater isolation includes closure of the feedwater regulating valves (FRV), the FRV bypass valves, the feedwater isolation valves (FIV), and trip of the steam generator feedwater pumps (SGFP). The response time for feedwater isolation by closure of the FRVs (the BF-19 valves) and the FRV bypass valves (the BF-40 valves) is 10 seconds. The response time for feedwater isolation by closure of the FIVs (the BF-13 valves) is 32 seconds. The response time for feedwater isolation by trip of the SGFPs is 7 seconds, not including pump coastdown time.

(9) Feedwater isolation only includes closure of the feedwater regulating valves (FRV) and the FRV bypass valve.

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Five containment cooling fans shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or two of the above required containment cooling fans inoperable, restore the inoperable cooling fan(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With three or more of the above required containment cooling fans inoperable, restore at least three cooling fans to OPERABLE status within 1 hour or be in at least HOT STANDBY WITHIN the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the remaining inoperable cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment cooling fan shall be demonstrated OPERABLE:

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SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 12 hours by:

1. Verifying the water level in each service water accumulator vessel is greater than or equal to 226 inches and less than or equal to 252 inches.

2. Verifying the temperature in each service water accumulator vessel is greater than or equal to 55°F and less than or equal to 95°F.

3. Verifying the nitrogen cover pressure in each service water accumulator vessel is greater than or equal to 135 psig and less than or equal to 160 psig.

b. At least once per 31 days by:

1. Starting (unless already operating) each fan from the control room in low speed.

2. Verifying that each fan operates for at least 15 minutes in low speed.

3. Verifying a cooling water flow rate of greater than or equal to ~~2550~~ 1300 gpm to each cooler.

c. At least once per 18 months by verifying that on a safety injection test signal:

1. Each fan starts automatically in low speed.

2. The automatic valves and dampers actuate to their correct positions and that the cooling water flow rate to each cooler is greater than or equal to ~~2550~~ 1300 gpm.

d. At least once per 18 months by verifying that on a loss of offsite power test signal, each service water accumulator vessel discharge valve response time is within limits.

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 Bases	Page
3/4.3.1 and 3/4.3.2	B 3/4 3-1a
3/4.6.2.3	B 3/4 6-3, 4

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

The verification of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety features actuation associated with each channel is completed within the time limit assumed in the safety analysis. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.). The Note 8 response times for feedwater isolation are based on WCAP-16503, "Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit (CFCU) Margin Recovery Project," Revision 3, (LCR S06-10). SGFP trip and FIV failure are credited in the containment analyses for LOCA and MSLB in case an FRV fails open.

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

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

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

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

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

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system, when operated in conjunction with the Containment Cooling System, ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

Normal plant operation and maintenance practices are not expected to trigger surveillance requirement 4.6.2.1.d. Only an unanticipated circumstance would initiate this surveillance, such as inadvertent spray actuation, a major configuration change, or a loss of foreign material control when working within the affected boundary of the system. If an activity occurred that presents the potential of creating nozzle blockage, an evaluation would be performed by the engineering organization to determine if the amount of nozzle blockage would impact the required design capabilities of the containment spray system. If the evaluation determines that the containment spray system would continue to perform its design basis function, then performance of the air or smoke flow test would not be required. If the evaluation cannot conclusively determine the impact to the containment spray system, then the air or smoke flow test would be performed to determine if any nozzle blockage has occurred.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

The surveillance requirements for the service water accumulator vessels ensure each tank contains sufficient water and nitrogen to maintain water filled, subcooled fluid conditions in three containment fan coil unit (CFCU) cooling loops in response to a loss of offsite power, without injecting nitrogen covergas into the containment fan coil unit loops assuming the most limiting single failure. The surveillance requirement for the discharge valve response time test ensures that on a loss of offsite power, each discharge valve actuates to the open position in accordance with the design to allow sufficient tank discharge into CFCU piping to maintain water filled, subcooled fluid conditions in three CFCU cooling loops, assuming the most limiting single failure.

Note: Shaded Text moved forward from page 3/4 6-4

The surveillance requirements for the CFCUs ensure sufficient SWS flow through each operating cooler to provide the minimum containment cooling as assumed by the containment response analysis for a design-basis LOCA or MSLB event. The surveillance flow rate is selected to ensure adequate heat removal (with no two-phase flow). The specified surveillance flow rate represents the total flow from both the CFCU coils and the CFCU motor-cooler.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The opening of locked or sealed closed containment isolation valves (penetration flow paths) on an intermittent basis under administrative control includes the following considerations: (1) stationing a dedicated individual, who is in constant communication with the control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

The main steam isolation valves (MSIVs) fulfill their containment isolation function as remote-manual containment isolation valves. The automatic closure of the MSIVs is not required for containment isolation due to having a closed system inside containment. The remote-manual containment isolation function of the MSIVs can be accomplished through either the use of the hydraulic operator or when the MSIV has been tested in accordance with surveillance requirement 4.7.1.5 the steam assist function can be credited.

Surveillance Requirement (SR) 4.6.3.1.3 only applies to the MS7 (Main Steam Drain) valves and the MS18 (Main Steam Bypass) valves. The MS167 (Main Steam Isolation) valves are tested for main steam isolation purposes by SR 4.7.1.5. For containment isolation purposes, the MS167s are tested as remote/manual valves pursuant to Specification 4.0.5.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment.

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

The verification of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety features actuation associated with each channel is completed within the time limit assumed in the safety analysis. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.). The Note 8 response times for feedwater isolation are based on WCAP-16503, "Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit (CFCU) Margin Recovery Project," Revision 3, (LCR S06-10). SGFP trip and FIV failure are credited in the containment analyses for LOCA and MSLB in case an FRV fails open.

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

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

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

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

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

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system, when operated in conjunction with the Containment Cooling System, ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

The containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

Normal plant operation and maintenance practices are not expected to trigger surveillance requirement 4.6.2.1.d. Only an unanticipated circumstance would initiate this surveillance, such as inadvertent spray actuation, a major configuration change, or a loss of foreign material control when working within the affected boundary of the system. If an activity occurred that presents the potential of creating nozzle blockage, an evaluation would be performed by the engineering organization to determine if the amount of nozzle blockage would impact the required design capabilities of the containment spray system. If the evaluation determines that the containment spray system would continue to perform its design basis function, then performance of the air or smoke flow test would not be required. If the evaluation cannot conclusively determine the impact to the containment spray system, then the air or smoke flow test would be performed to determine if any nozzle blockage has occurred.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

The surveillance requirements for the service water accumulator vessels ensure each tank contains sufficient water and nitrogen to maintain water filled, subcooled fluid conditions in three containment fan coil unit (CFCU) cooling loops in response to a loss of offsite power, without injecting nitrogen covergas into the containment fan coil unit loops assuming the most limiting single failure. The surveillance requirement for the discharge valve response time test ensures that on a loss of offsite power, each discharge valve actuates to the open position in accordance with the design to allow sufficient tank discharge into CFCU piping to maintain water filled, subcooled fluid conditions in three CFCU cooling loops, assuming the most limiting single failure.

Note: Shaded Text moved forward from page 3/4 6-4

The surveillance requirements for the CFCUs ensure sufficient SWS flow through each operating cooler to provide the minimum containment cooling as assumed by the containment response analysis for a design-basis LOCA or MSLB event. The surveillance flow rate is selected to ensure adequate heat removal (with no two-phase flow). The specified surveillance flow rate represents the total flow from both the CFCU coils and the CFCU motor-cooler

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The opening of locked or sealed closed containment isolation valves (penetration flow paths) on an intermittent basis under administrative control includes the following considerations: (1) stationing a dedicated individual, who is in constant communication with the control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring that the environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

The main steam isolation valves (MSIVs) fulfill their containment isolation function as remote-manual containment isolation valves. The automatic closure of the MSIVs is not required for containment isolation due to having a closed system inside containment. The remote-manual containment isolation function of the MSIVs can be accomplished through either the use of the hydraulic operator or when the MSIV has been tested in accordance with surveillance requirement 4.7.1.5 the steam assist closure function can be credited.

Surveillance Requirement (SR) 4.6.3.3 only applies to the MS7 (Main Steam Drain) valves and the MS18 (Main Steam Bypass) valves. The MS167 (Main Steam Isolation) valves are tested for main steam isolation purposes by SR 4.7.1.5. For containment isolation purposes, the MS167s are tested as remote/manual valves pursuant to Specification 4.0.5.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.