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10 CFR 50.90

LR-N07-0222

United States. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

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: **RESPONSE TO RAI #3 ON LCR S06-10 (TAC Nos. MD4843 & 4844) STEAM GENERATOR FEEDWATER PUMP TRIP, FEEDWATER ISOLATION VALVE RESPONSE TIME TESTING and CONTAINMENT COOLING SYSTEM**

References:

(1) Letter from PSEG to NRC: "License Change Request for S06-10, Steam Generator Feedwater Pump Trip, Feedwater Isolation Valve Response Time Testing and Containment Cooling System, Salem Nuclear Generating Station, Units 1 and 2, Facility Operating Licenses DPR-70 and DPR-75, Docket Nos. 50-272 and 50-311", dated March 16, 2007

(2) Letter from NRC to PSEG: "Salem Nuclear Generating Station, Units 1 and 2, Request for Additional Information, Amendment Request Re: Steam Generator Feedwater Pump Trip, Feedwater Isolation Valve Closure Response Times, and Containment Fan Coil Unit Cooling Water Flow Rate (TAC Nos. MD4843 & 4844)", dated August 28, 2007

(3) Letter from PSEG to NRC: "Response to RAI#1 and RAI#2 on License Change Request for S06-10, Steam Generator Feedwater Pump Trip, Feedwater Isolation Valve Response Time Testing and Containment Cooling System, Salem Nuclear Generating Station, Units 1 and 2, Facility Operating Licenses DPR-70 and DPR-75, Docket Nos. 50-272 and 50-311", dated August 30, 2007

In Reference 1, PSEG Nuclear LLC (PSEG) submitted License Change Request (LCR) S06-07 to amend the Technical Specifications (TS) for the Salem Nuclear Generating Station, Units 1 and 2 (Reference 1). LCR S06-10 entails (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

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Containment Fan Cooler Units (CFCUs), allowing a reduction in the required Service Water (SW) flow to the CFCUs.

The NRC has provided PSEG three Request for Additional Information (RAI) on LCR S06-10; these three RAIs were collectively provided via Reference 2. In Reference 3, PSEG submitted the response to RAI#1 and RAI#2 (Questions EMCB-1, SBPB-1, SBPB-2, and SBPB-3).

On August 14th, 2007 and August 22nd, 2007, PSEG and the NRC discussed RAI#3 via teleconference to provide additional clarification. The response to RAI#3 (Questions EEEB-1 through EEEB-9, SCVB-1 through SCVB-7, and SRXB-1) is provided as an attachment to this submittal. Additional proposed changes to the TS and TS Bases are also provided as attachments to this submittal.

In accordance with 10CFR50.91(b)(1), a copy of this letter has been sent to the State of New Jersey.

PSEG has evaluated the additional information provided in Attachment 1 in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined there is no impact to the no significant hazards consideration provided in Reference 1. There is also no impact to the 10 CFR 51.22(c)(9) environmental assessment provided in Reference 1.

If you have any questions or require additional information, please do not hesitate to contact Mr. Steve Mannon at (856) 339-1129.

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

Executed on 9/14/07
(Date)

Sincerely,

RC

Robert C. Braun
Site Vice President
Salem Generating Station

Attachments (3)

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REQUEST FOR ADDITIONAL INFORMATION #3

REGARDING PROPOSED LICENSE AMENDMENT

STEAM GENERATOR FEEDWATER PUMP TRIP,

FEEDWATER ISOLATION VALVE CLOSURE RESPONSE TIMES,

AND CONTAINMENT FAN COIL UNIT COOLING WATER FLOW RATE

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

By letter dated March 16, 2007, PSEG Nuclear LLC (PSEG or the licensee) submitted an amendment request for Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. The proposed amendment would add new Technical Specification (TS) requirements for the response times associated with a steam generator feedwater pump (SGFP) trip and feedwater isolation valve (FIV) closure. The amendment would also revise the TS requirements for the containment fan cooler unit (CFCU) cooling water flow rate. These changes are associated with a revised containment response analysis that credits a SGFP trip and FIV closure (on a feedwater regulator valve failure) to reduce the mass/energy release to the containment during a main steam line break (MSLB). The containment analysis also credits a reduced heat removal capability for the CFCUs, allowing a reduction in the required service water (SW) flow to the CFCUs.

The NRC staff has reviewed the information the licensee provided that supports the proposed amendment and would like to discuss the following issues to clarify the submittal.

EEEB-1: Page 4 of Attachment 1 to the application states that the duration of the analyses for the new LOCA cases were extended to approximately 120 days to support the environmental qualification (EQ) bases for the critical components. Please confirm that all EQ equipment are qualified for the new LOCA cases for 120 days.

RESPONSE

All EQ equipment is qualified for the new LOCA cases for 120 days.

WCAP-16503 analyzed long-term LOCA mass and energy release break cases to support the 120 day Post Accident Operability Period (PAOP) utilized in the Salem Station EQ Program PAOP analysis. The EQ Program PAOP has not changed and has always been evaluated to 120 days for EQ equipment inside Salem containment.

The use of the term "critical equipment" is misleading, as it is a term normally used in the Performance Centered Maintenance (Maintenance Rule) program. The term "critical equipment" should be replaced with the term "safety-related equipment", indicating all EQ equipment. (refer also to EEEB-9 response)

Additional Clarification on EQPro software (as requested during the August 14, 2007 conference call between the NRC and PSEG)

Salem Station maintains EQ Binders on each equipment type that constitutes the auditable documentation required by 10 CFR 50.49 for the EQ Program. In October of 2005, an EQ Program Optimization Project was initiated to transition the original hardcopy EQ Binders format, with tabular sections and hand written data, into an electronic, word searchable, streamlined data platform using a software application called *EQPro*.

EQPro is a database that electronically administers EQ data while providing an efficient means for information access, retrieval, and update. The *EQPro* software includes automated functions for performing EQ calculations for qualified life and post-accident operability. The software has modules dedicated to the three main categories of EQ data:

1. Environmental Parameters
2. Plant Equipment Identification
3. EQ Binder

The resultant *EQPro* EQ Binder is a specially designed database report that provides the entire EQ evaluation in a single document.

EQPro was developed in accordance with a Quality Assurance Program, including validation & verification (V&V). Development meets the requirements of ASME NQA-2a-1990, Part 2.7. For Salem Station, *EQPro* constitutes a developed software product with quality control requirements specified by applicable PSEG procedures. The updated EQ analysis in support of the Replacement Steam Generator (RSG) Project is being accomplished concurrent with the *EQPro* EQ Binder Optimization Project.

EEEB-2: Section 6.5 of Engineering Evaluation S-C-CBV-MEE-1982 (Enclosure 2 to the application) indicates that Salem Unit 2 bounding profiles are used to qualify safety-related equipment in containment. Please confirm that the Unit 1 profiles are bounded by the Unit 2 bounding temperature and pressure profiles.

RESPONSE

The Unit 1 EQ accident profiles are bounded by the Unit 2 RSG temperature and pressure profiles.

RSG Unit 2 and Unit 1 bounding accident case analysis were originally developed in 2005. Subsequent analyses were performed in 2006 and a final bounding EQ temperature accident profile was completed in May 2007. All EQ equipment was reviewed against the May 2007 bounding EQ temperature profile.

In May 2007, PSEG finalized the RSG Containment temperature EQ profile inputs into the *EQPro* software utilizing worst case data provided in Westinghouse WCAP-16503, Rev 3, dated February 2007, which contained the final temperature and pressure cases that would be used to develop the Containment EQ Temperature Profile and qualify the EQ equipment inside containment to a PAOT of 120 days. WCAP-16503 superseded previous temperature and pressure data in Calc Note CN-CRA-04-41. This final EQ temperature profile is documented in Calculation 630-005-DC1, dated May 2007, "Salem Containment Updated Composite Temperature & Pressure Accident Profiles for Environmental Qualification for CFCU Margin Recovery and Steam Generator Replacement". This EQ Temperature Profile is a composite of all Westinghouse Design Basis Event (DBE) cases and envelopes the Unit 2 RSG and the Unit 1 and 2 Original Steam Generator (OSG) temperature profiles.

The original DBE peak pressure (47 psig) and peak temperature (351°F) remain bounding and all EQ equipment remains qualified for the 120 day PAOP.

In all of Westinghouse analyzed pressure case studies, the pressure effect caused by the RSGs was determined to be bounded by the current peak design basis pressure of 47 psig, and thus no further analysis was required to qualify the EQ equipment.

Additional Clarification Response - Outside Containment Impact (as requested during the August 14, 2007 conference call between the NRC and PSEG)

Westinghouse evaluated the thermal response of safety-related equipment outside of containment in the Outboard Main Steam Penetration Access Area (OAA)) in LTR-EMPE-05-179 (VTD 328362), dated June 2005, titled, "Justification for Use of Class 1E Equipment in FU/MRP and RSG SLB Environments in the OAA for Salem Units 1 and 2". In this evaluation, Westinghouse considered the thermal response of Safety-Related equipment in the OAA (FU/MRP is the Fuel Upgrade/Margin Recovery Project). This Westinghouse analysis was performed to confirm that the EQ equipment located in the OAA will continue to be qualified for the new transients resulting from the postulated MSLB based on the Unit 2 RSG program. The evaluation compares the environmental qualification EQ test data to the OAA conditions resulting from the RSG project accident conditions and the Fuel Upgrade/Margin recovery Program (FU/MRP). Based on these results, Westinghouse indicates that the Unit 2 OAA temperature response is more severe than Unit 1 and thus the Unit 2 OAA analysis is bounding for both Salem Units 1 and 2. Westinghouse documented the EQ equipment temperature response in Calc Note CN-CRA-04-51, titled, "Salem Unit 2 Outboard Penetration Access Area Temperature Response to a Steam line Break for the RSG Project".

The Westinghouse OAA thermal response data was input into the *EQPro* program database to evaluate the temperature conditions in support of qualification of the EQ equipment in the OAA areas. This evaluation was performed in June 2007 and is a supporting document for acceptance of the Westinghouse analysis. This evaluation is documented in Calculation 630-005-DC2, dated June 2007, and titled, "Justification for Applying the Thermal Response of OAA Equipment in Support of Environmental Qualification". Based on the results of this calculation all EQ equipment in the OAA areas remains qualified to the new transients resulting from the RSG Project.

EEEB-3: Section 6.5 of Enclosure 2 of the application states that the "EQPro" profile has been used to evaluate the EQ equipment for an unrelated EQ Program update. Please clarify what this statement means.

RESPONSE

The reference in Section 6.5 of Enclosure 2 to an "unrelated project" refers to the ongoing EQ Program Binder Optimization Project using the *EQPro* software.

As discussed in the response to EEEB-1, the EQ Program Binder Optimization Project was initiated in October 2005 and is still in progress. The updated EQ analysis in support of the RSG Project is being accomplished concurrently with the *EQPro* EQ Binder Optimization Project.

EEEB-4: Please clarify whether "EQPro" input curves are used as bounding profiles for EQ.

RESPONSE

The *EQPro* input curves were used to develop the bounding Design Basis Event (DBE) temperature profile (EQ Profile) for the qualification of EQ equipment.

Finalized RSG Containment temperature and pressure EQ profiles were generated by the *EQPro* database utilizing data provided in Westinghouse WCAP-16503, Rev 3, dated February 2007. This final EQ temperature profile is documented in Calculation 630-005-DC1, dated May 2007, "Salem Containment Updated Composite Temperature & Pressure Accident Profiles for Environmental Qualification for CFCU Margin Recovery and Steam Generator Replacement". This EQ Temperature Profile is a 120 day composite of all Westinghouse Design Basis Accident (DBE) cases and envelopes the Unit 2 RSG and the Unit 1 and 2 OSG temperature profiles as indicated in the Salem Environmental Design Criteria (EDC), document number S-C-ZZ-SDC-1419.

EEEB-5: Section 6.5 of Enclosure 2 of the application states that "The EQ analyses of critical equipment are based on composite curves that envelope the estimated temperature, pressure and radiation environments during a design basis event. These composite curves are defined in the Salem Environmental Design Criteria EDC (Reference 15)." Please confirm that these composite curves are bounded by EQPro input curves. If not, then explain the impact of these composite curves on the EQ program.

RESPONSE

The composite curves are bounded by the *EQPro* input curves.

The *EQPro* EQ Temperature Profile is a 120 day composite of all Westinghouse DBE cases and the Original Steam Generator (OSG) temperature DBE profile. The new EQ temperature profile is documented in Calculation 630-005-DC1, "Salem Containment Updated Composite Temperature & Pressure Accident Profiles for Environmental Qualification for CFCU Margin Recovery and Steam Generator Replacement".

EEEB-6: Section 6.5 of Enclosure 2 of the application states that the containment temperatures exceed the current analysis of record (AOR). Please provide detailed explanation on how the EQ equipment are qualified where the AOR is exceeded.

RESPONSE

Section 6.5 of Enclosure 2 of the application states that the existing analysis of record (AOR) is exceeded in some areas, but goes on to identify the specific areas, which are before (ramp-up) and after (ramp-down) the peak temperature of 351°F. As discussed in the three cases presented in the application, at no time is the containment maximum design temperature of 351°F exceeded. In all areas where the current AOR was exceeded, before and after the peak, the revised EQ Composite Temperature Profile has been increased and the EQ equipment reanalyzed and qualified to the revised conditions. The method used to qualify the equipment is discussed below.

The EQ equipment are qualified to the new EQ temperature profile documented in Calculation 630-005-DC1, dated May 2007, "Salem Containment Updated Composite Temperature & Pressure Accident Profiles for Environmental Qualification for CFCU Margin Recovery and Steam Generator Replacement". This EQ Temperature Profile is a 120 day composite of all Westinghouse Design Basis Accident (DBE) worst cases documented in WCAP-16503, Rev 3, and envelopes the Unit 2 RSG and the Unit 1 and 2 OSG temperature profiles in all areas that exceed the current analysis of record.

The EQ equipment was qualified by direct comparison of equipment test data to the new EQ temperature profile and by utilizing the Arrhenius aging calculation. Post accident operability was evaluated by extrapolating vendor EQ equipment test data against the new EQ temperature profile documented in Calculation 630-005-DC1. This feature is provided as part of the *EQPRO* software. Vendor qualification accident test data were taken from existing EQ Binder vendor test reports utilizing the most limiting activation energies. The first 24 hours of each of the vendor test data reports was compared directly to the new Salem EQ profile and the remainder of the 120 PAOT was then evaluated against vendor test data utilizing Arrhenius aging methodology, extrapolating each equipment type to the new EQ temperature profile over the remainder of post accident period. Results indicated that all EQ equipment inside containment has greater than 10% test margin and remained qualified to the worst case Westinghouse pipe break cases analyzed for the RSG, as documented in Calculation 630-005-DC1.

EEEB-7: Please confirm that the radiation dose for the proposed changes remain unchanged. If not, then provide its impact on the EQ Program.

RESPONSE

The radiation dose for EQ program remains unchanged and the EQ equipment in containment did not require a re-analysis for the RSG project.

The radiation dose consequences for the RSG project were analyzed in Calculation S-C-ZZ-MDC-1945, "Post LOCA, EAB, LPZ & CR Doses – Alternate Source Term (AST), and in conjunction with Calculation S-C-ZZ-MDC-2008, "Radiological Impact of the AST on the

Salem Unit 1 & 2 EQ Program". The post LOCA 120 day gamma dose was evaluated for the safety related electrical equipment exposed to the LOCA event and the post LOCA recirculation piping. It was determined that the existing conservatism in the Salem total integrated dose (TID) source term for the post LOCA sump water compensates for the increase in the 120 day integrated gamma dose due to the increased cesium in the AST (Calculation S-C-ZZ-MDC-2008) and can be applied to the RSG analysis (Calculation S-C-ZZ-MDC-1945) .

The Steam Generator Replacement DCP 80083522, Section 4.1.24.8, indicates that the RCS volume will increase slightly (approximately 3.2%). This increase in RCS volume has negligible impact on post-LOCA radiological consequences. For example, assuming the mass is increased by 3.2%, the overall calculated dose is 4.12E-04 rem TEDE. The results also indicate that the instantaneous release of fission products is a negligible contributor to the overall radiological consequence. The calculation S-C-ZZ-MDC-1945 also assessed the impact of the updated sump water temperature histories on iodine flashing factors and iodine release rates and found that the net effect is that doses will decrease slightly (less than 1%).

EEEB-8: If EQ equipment are requalified or replaced due to the proposed changes, please provide the details on requalification or replacement of the EQ equipment and confirm that the EQ and maintenance programs reflect these changes.

RESPONSE

No EQ equipment has been identified to be requalified or replaced due to the RSG project.

EEEB-9: The application states that the EQ analysis is performed for "critical equipment." Please define/clarify this term.

RESPONSE

The term "critical" equipment is a misused term. The correct term is to state that the EQ analysis is performed for "safety-related" equipment, indicating all EQ equipment.

SCVB-1: It was stated in several places of the license change request that the AOR for the single failure scenario of the faulted loop feedwater regulating valve (FRV) failing open is overly conservative as it assumed full feedwater (FW) flow to the faulted steam generator (SG) for 32 seconds, until the FIV is 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. In addition, the revised modeling assumes that during the SGFP coastdown, FW flow will decrease linearly to the flowrate provided by the condensate pumps through freewheeling SGFPs. The revised modeling has broken down the 32 second closing time of the FIV and the 14 second time for the SGFP to come to a stop from freewheeling as follows:

FIV: Two-second electronic time 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.

SGFP: Seven seconds for tripping of the SGFPs (instrument and mechanical delays), and seven seconds for coast down.

The NRC staff requests the following clarifications:

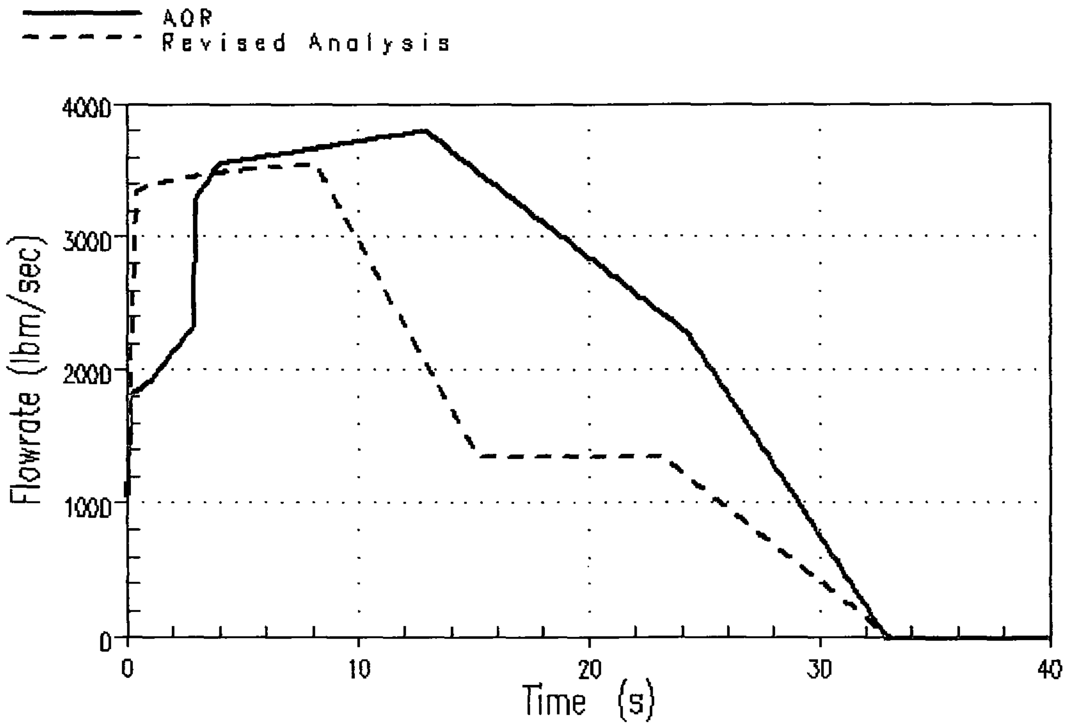
a) How does the FW flow to the faulted SG differ between the AOR and the revised model?

RESPONSE (SCVB-1a)

The AOR does not assume full feedwater flow to the faulted SG for 32 seconds. The overly conservative assumption of the AOR was that the trip of the main feedwater (MFW) pumps was not considered. However, the AOR had complex assumptions associated with the effect of valves. It assumed 1) a ramp opening of the feedwater control valve (FCV) on the faulted loop at the beginning of the event lasting until 3 seconds after the SI setpoint is reached, 2) the sudden closure of the intact loops' FCVs at 2 seconds after the SI setpoint is reached, and 3) a decrease in the MFW flowrate due to the closure of the feedwater isolation valve (FIV) over the last 20 seconds of the 30-second stroke time of the FIV. These assumptions have been in the AOR since at least the 1993 Fuel Upgrade / Margin Recovery Project.

The MFW flow to the faulted SG in the latest Salem analysis has credited the benefit of the MFW pump trip, but is based on more conservative assumptions (i.e., more limiting for the analysis) than the AOR relative to the modeling of valve positions and their effect on the MFW flowrate. The revised analysis conservatively assumes that the FCV on the faulted loop immediately fully opens in response to the steamline break, while the FCVs on the intact loops are assumed to immediately close. The revised analysis also credits a flow reduction due to the closure of the FIV during only the last 10 seconds of the 30-second stroke time.

This discussion pertains to the SLB cases with a single failure of the FCV (failed open) on the faulted loop. It also only addresses the addition of MFW due to pumped flow; there is additional MFW added to the faulted SG due to flashing of the water in the unisolable feedline volume between the SG and the FIV, after the FIV closes. The plot below shows a comparison of the pumped MFW flowrate delivered to the faulted SG for the AOR compared to the revised analysis, using an example of a 1.4 ft^2 double-ended rupture SLB on Unit 2 initiated from full power.



b) Provide, in quantitative terms, flow considered in the AOR for the entire 32 seconds FIV closure and a breakdown of flow in the revised model during the seven seconds of the SGFP trip, seven seconds of SGFP coast down, the time remaining (eight seconds) prior to the linear flowrate reduction, and the final 10 seconds of the FIV stroke during which flowrate is reduced linearly.

RESPONSE (SCVB-1b)

The integrated pumped MFW flow delivered to the faulted SG in the AOR is approximately 87,000 lbm.

In the revised analysis the integrated MFW flow is:

- 27,700 lbm at 8.1 seconds, when the MFW pump coastdown starts,
- 45,100 lbm at 15.1 seconds when the flow coastdown of the MFW pump stops,
- 55,900 lbm at 23.1 seconds when the reduction due to the FIV closure starts,
- 62,900 lbm when the FIV is fully closed at 33.1 seconds.

SCVB-2: The proposed amendment includes: (1) new TS requirements for the response times associated with a SGFP trip and FIV closure; and (2) revised TS requirements for the CFCU cooling water flow rate. However, new TS requirements for SGFP coastdown time was not included in the proposed amendment. Justification for non-inclusion of the pump coast down in the TS was provided in Section 5.4 of Engineering Evaluation S-C-CBV-MEE-1982 (Enclosure 2 to the application). It is stated in the evaluation that a survey of other plants that have credited FW pump coastdown in their MSLB containment response analysis has identified a range of values between five and ten seconds and that none of the plants have included the coastdown values in their TSs. Additional pump coastdown information from two plants, Indian Point and Diablo Canyon, was provided in the Engineering Evaluation, including a comparison of turbine/pump sets between Diablo Canyon and Salem.

The NRC staff requests responses to the following questions:

a) What is the context in which the pump coastdown information was used in the analysis pertaining to other plants?

RESPONSE (SCVB-2a)

S-C-CBV-MEE-1982 included results of a survey of other plants that have credited SGFPs coastdown in the MSLB containment response analysis to demonstrate: (1) Salem is not the first plant to credit SGFP coastdown; and (2) the coastdown parameters used in the Salem analysis are consistent with those used for other plants.

Detailed comparisons are made to Diablo Canyon Power Plant (DCPP) due to the similarity of the plants and SGFP/turbine drive characteristics. The DCPP analyses were performed as part of the SG replacement, like Salem. The DCPP analysis assumes a 5-second signal/hardware processing delay and a 5 second SGFP coastdown time. The 5-second signal delay is comprised of a 2 second signal processing delay, a 1 second slave relay actuation, a 1 second SGFP steam stop stroke time and 1 second of "hardware" margin. The DCPP surveillance testing requires the total SGFP trip function response time of less than 9 seconds, including signal-processing time, pump trip, and pump coastdown. Of this time, an administrative limit of 5 seconds is used for the hardware related response time, leaving 4 seconds for pump coastdown.

The DCPP accident analysis assumption of 5 seconds of SGFP coastdown bounds the 4-second coastdown. The NRC has accepted these assumptions which are documented in the NRC Safety Evaluation Report for DCPP Technical Specification Amendment 140, dated February 22, 2000.

The Salem SGFPs and turbine drives are overall similar to those at DCPP. The revised Salem analysis allows for a longer coastdown, which is conservative.

b) Was it used for a similar purpose as for Salem (i.e., to perform a refined analysis in order to justify a significant reduction in cooling water flow to the CFCUs)?

RESPONSE (SCVB-2b)

The proposed change in Salem SGFP timing surveillance criteria is being requested to support the Salem Unit 2 Steam Generator Replacement Project. The SGFP trip assumptions affect the amount of feedwater mass that is pumped into the faulted steam generator which impacts the peak containment pressure and time. This impact is considered to be in the "near term" duration during an accident. The proposed CFCU cooling water flow reduction modification primarily affects the long term containment cooling function and therefore the environmental qualification temperature profile for the containment.

Diablo Canyon did not use the SGFP timing changes to justify a reduction in cooling water flow to the CFCUs. The change in SGFP timing accident analysis assumptions was undertaken to support the accident analysis assumptions for the DCPP Replacement Steam Generator Project.

c) What is the sensitivity of the Salem analysis if the SGFP takes longer than 7 seconds to coastdown?

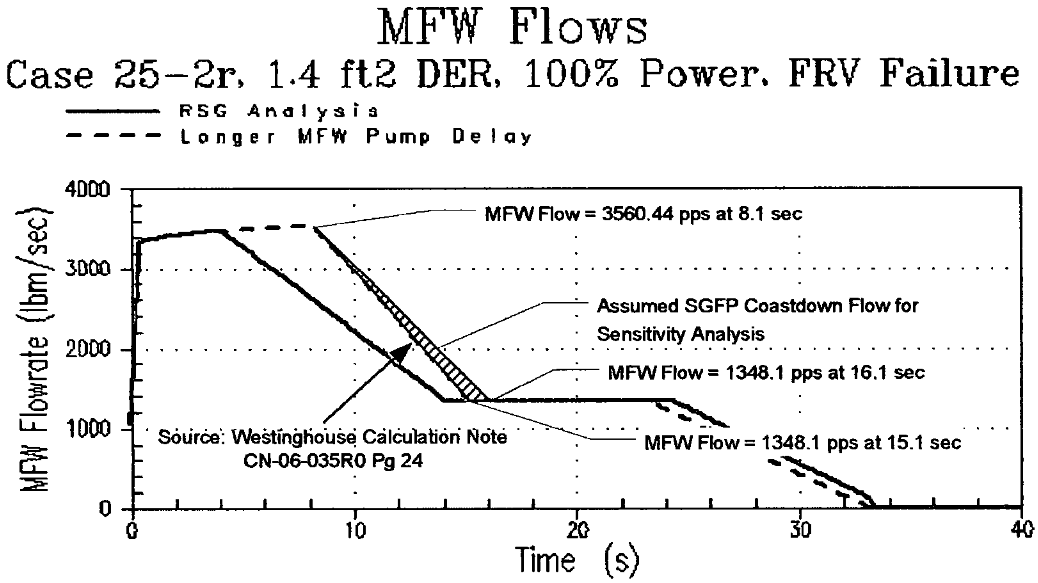
RESPONSE (SCVB-2c)

The figure below is a markup of a feedwater flow plot from Westinghouse CN-CRA-06-035 which shows the sensitivity of the feedwater flow to the pump coastdown time. In the plot, the SGFP trip occurs at 8.1 seconds into the transient. The Main Feedwater (MFW) flow decreases from a peak of 3560.4 lb/sec (pps) to 1348.1 lb/sec following pump coastdown. Neglecting the change in SG pressure over the interval of the increased SGFP coastdown (15.1 to 16.1 seconds) time will have minimal impact. The integrated MFW mass will increase by approximately 1100 lbm using the integrated hatched area from the plot below. The sensitivity of the containment peak pressure to a change in integrated mass flow is approximately 1.52E-4 psig/lbm based on an extrapolation of the CN-CRA-06-035 calculated containment peak pressures and integrated feedwater mass. The result is that the 1 second increase in the SGFP coastdown time would increase the integrated MFW mass by approximately 1100 lbm and would increase the containment peak pressure by 0.17 psig. The Unit 2 containment peak pressure for this scenario would be approximately 44.27 psig, which is less than the containment design pressure of 47.0 psig and would be acceptable.

For comparison, the limiting Unit 2 RSG case is a 1.4 ft^2 Double Ended Rupture (DER) at 30% Power with a containment safeguards train failure and has a peak containment pressure of 45.6 psig. The change in SGFP coastdown characteristics only affects those containment analysis transients that assume a feedwater regulating valve failure.

Therefore, an increase in the coastdown from 7 seconds to 8 seconds has a minimal impact on the integrated mass flow and containment peak pressure.

SGFP Coastdown Assumption Sensitivity



SCVB-3 Section 5.5 of Engineering Evaluation S-C-CBV-MEE-1982 provides a discussion of the flow characteristics during FIV closure. This section recognizes that FW flow to the faulted SG may be higher than what was considered in the WCAP analysis, however, it provides a discussion as to why the results of the analysis will not be significantly affected. The NRC staff has the following observations and is requesting additional clarifications:

a) The 1st and 2nd paragraphs seems to be contradicting. In reference to the WCAP analysis, the 1st paragraph (page 12 of 54) states that "...instead of decreasing the flow over the full 30 second design basis stroke of the valve, the FIV is only credited to close with a linear flow ramp over the last 10 seconds of the 30-second stroke time. The WCAP analysis assumes a full open valve resistance coefficient for the first 20 seconds of the stroke, even though the valve will have completed about 66% of its closing stroke (i.e., valve will only be approximately 33% open when the model begins the linear decrease in flow)." However, in the 2nd paragraph (page 13 of 54) it states that "[i]n general, gate valves do not significantly affect system flow until they are less than 50% open. Pages 44 thru 50 of the original BF13 MOV Calc S-1-CN-MDC-0881 Sheet 001 (Reference 13) evaluates the effect of BF13 closure on feedwater flow and concludes that the linear flow decrease assumption during the final 5% of the stroke is not valid for the BF-13 operating conditions during a MSLB event. In particular, pages 49 and 50 of MDC-0881 (Reference 13) identify high flow through BF13 (>5000 gpm) even down to 5% open because of the choked flow effects - i.e., it specifically states the valve Cv does not change in a linear manner." Clarify the intent of the above statements and how they can be construed as supporting the intent.

RESPONSE (SCVB-3a)

These two paragraphs only provide background information.

- The 1st paragraph summarizes the WCAP analysis assumptions.
- The 2nd paragraph makes the observation that, in general, the hydraulic resistance of gate valves starts to increase in a linear fashion when the valve flow area is less than 50% except for the final 5%, which may result in flow choking at the gate valve due to the high pressure drops.

Assuming full flow for the first two-thirds of the closing stroke and linearly decreasing flow over the final one-third stroke is appropriate as it delays the reduction in flow and ignores the flow limitation when choking occurs.

b) Figure 5.5-1 of Engineering Evaluation S-C-CBV-MEE-1982 shows a plot of the FW mass flow rate as a function of the FIV stroke time. In a discussion related to FW flow in this figure (page 13 of 54), it was stated that "Note: Case 8 represented feedwater flow with 0 psia SG pressure and MFW pumps off (Condensate pumps only)." The NRC staff requests confirmation that the actual flow used in the WCAP analysis is higher in the first seven seconds due to the fact that the SGFP would not have tripped until that time, as well as the next seven seconds when the SGFP is coasting down.

RESPONSE (SCVB-3b)

The actual flow used in the WCAP analysis is higher in the first seven seconds due to the fact that the SGFP would not have tripped until that time, as well as the next seven seconds when the SGFP is coasting down. The transient feedwater flow response, and associated plot, is provided in the response to NRC RAI Question SCVB-1.a.

c) In the 1st paragraph of page 14 of 54 of Engineering Evaluation S-C-CBV-MEE-1982, it states that "[t]he as-tested BF-13 stroke time of 26 seconds (Reference 13) provides additional rationale for concluding that the actual feedwater mass injected into a faulted steam generator will be less than the assumed value from WCAP-16503." Since: (1) the proposed TS includes a 32 second response time (consisting of 2 seconds of electronic delay and 30 seconds of stroke time); (2) the associated surveillance procedure will only verify the 30 second stroke time; and (3) the 26 second stroke time is based on a test performed in 1995; the NRC staff has concerns regarding the impact on the licensee's analyses, if the actual stroke time is greater than 26 seconds. Please clarify.

RESPONSE (SCVB-3c)

The revised containment analysis in WCAP-16503 uses a 30-second stroke time for the BF-13 valves. This value is consistent with the TS response time testing.

The BF-13 valves are stroked per S1(2).OP-ST.MS-0002. The table below summarizes results from the most recent stroke time tests for the BF-13s at Unit 2. As shown, the most recent stroke time results are within a fraction of a second of the baseline value. Further, all values are within the acceptance criteria. The Unit 1 values are similar.

Valve	Reference Stroke Time (sec)	Date of Reference Value	Surveillance Test Acceptance Criteria (secs)	Latest Stroke Time Results (sec)	Last Surveillance Test
21BF13	26.5	12/19/94	22.5-29.0	26.4	10/25/06
22BF13	26.0	12/19/94	22.1-29.0	26.4	10/25/06
23BF13	26.7	10/24/06	22.7-29.0	26.7	10/24/06
24BF13	25.8	12/19/94	21.9-29.0	26.6	10/24/06

The above requirements provide assurance that the as-found BF-13 stroke times will be less than the 30-second stroke times assumed in WCAP-16503. Surveillance testing results remain in the 26 seconds range, and must be less than 29 seconds to meet the acceptance criteria.

d) In the 1st paragraph of page 14 of 54 of Engineering Evaluation S-C-CBV-MEE-1982, it states that "[c]onsidering the conservatism in flow assumed during the first 20 seconds of the valve stroke, assuming a linear flow reduction in the total system flow over the final 10 seconds of BF-13 valve stroke time is considered to be a reasonable assumption." What is the conservatism in the first 20 seconds, considering that gate valves do not significantly affect system flow until they are less than 50% open?

RESPONSE (SCVB-3d)

The conservatism in the assumption in only reducing feedwater flow over the last 10 seconds of BF-13 valve stroke time is that full flow is assumed to occur until the valve is 33% open (10 seconds/30 second full stroke time) rather than 50% open. A typical gate valve (See Idelchik "Handbook of Hydraulic Resistance" pg 360) hydraulic loss coefficient starts to provide some hydraulic resistance approximately 50% open.

SCVB-4: In Section 6.1 of Engineering Evaluation S-C-CBV-MEE-1982 (page 18 of 54), it states that "[t]he revised peak containment temperature is 349.6 °F vs 348.2 °F (difference of +1.4 °F)." The NRC staff requests clarification of the discrepancy between the referenced numbers and those given in Table 6.1-1 of the Engineering Evaluation.

RESPONSE

PSEG acknowledges the discrepancy in S-C-CBV-MEE-1982. The peak containment temperature for an MSLB in the AOR is 351°F as stated in the Table 6.1-1. PSEG calculations 6S0-1800, 6S0-2027, and 6S0-2028 evaluate the containment liner and anchors, demonstrating the adequacy of the containment liner for the current AOR peak temperature of 351°F. Since the peak temperature in the AOR bounds the revised analysis (WCAP-16503, Rev.3), there is no impact on the containment liner and liner anchors. S-C-CBV-MEE-1982 will be revised to provide clarity.

SCVB-5: Engineering Evaluation S-C-CBV-MEE-1982 Section 7.2 (page 40 of 54) acknowledges that increasing the normal SW flowrate to the CFCUs from 700 gallons per minute (gpm) to 1300 gpm also has the additional advantage of improving containment cooling for operations during the summer months. Has any analysis been conducted to quantify the improved cooling in terms of normal containment temperatures? If so, what are the results?

RESPONSE

No analysis has been performed to quantify the increased cooling during normal operation with the increased SW flow to the CFCUs. Increasing the SW flow will increase the tubeside heat transfer coefficient. However, the benefit of increased flow during normal operation is small (less than 1°F) because the dominant thermal resistance term in the overall heat transfer coefficient is forced convection to the relatively dry containment air. Formal quantification of the benefit was deemed unnecessary because there is no negative impact in heat removal capability during normal operation, and therefore no impact on the ability to comply with the current Technical Specification 3.6.1.5 maximum containment temperature limit of 120°F.

SCVB-6: In Attachment 2, "Technical Specification Pages with Proposed Changes," there are some inconsistencies in how the notes are called out in Table 3.3-5. In some cases the note number is in parentheses, in some cases the note number is in parentheses in superscript and in some cases the table says "Note x." The licensee may want to consider making these consistent.

RESPONSE

PSEG concurs that Table 3.3-5 would be improved by making the notation consistent; the notes will be made consistent by depicting the note number in parenthesis in superscript. The required changes are reflected (shaded) in the marked-up TS pages included with this submittal (Attachment 2). Note that recently approved (i.e., after the original submittal of LCR S06-10) Salem Amendments 283 & 266 relocated the instrument response time limits for the reactor trip system and engineered safety features actuation system from TS Tables 3.3-2 and 3.3-5 to the Salem Updated Final Safety Analysis Report.

SCVB-7: In Attachment 3, "Proposed Changes to TS Bases Pages," on page B 3/4 3-1a (for both Unit 1 and Unit 2), it states that "SGFP trip and **FIV failure** are credited in the containment analysis for LOCA and MSLB in case an FRV fails open." It is suggested that "FIV failure" be changed to "FIV closure" or to "FIV isolation."

RESPONSE

PSEG concurs with the suggestion; the Bases pages will be changed as indicated (to "FIV closure"). See shaded changes in Attachment 3 to this submittal.

SRXB-1: Please discuss the steam generator tube rupture analysis for the replacement steam generators, or describe why such analysis is not needed, or that it is bounded by the analysis of record.

RESPONSE

The worst-case radiological results from the SGTR case for the RSG remain bounded by the AOR offsite and control room radiological consequences evaluation. Calculation S-C-ZZ-MDC-1949 (EAB, LPZ, & CR Doses – Steam Generator Tube Rupture (SGTR) Accident – AST) documents the dose calculation for the existing steam generators for both Units and the replacement steam generators for Unit 2. The calculation concluded that doses are within allowable limits.

A steam generator tube rupture (SGTR) thermal-hydraulic analysis for calculation of the radiological consequences has been performed for the Unit 2 Replacement Steam Generator (RSG). It is documented in WCAP-16444.

- The assumptions in WCAP-16444 are consistent with the SGTR analysis presented in Section 15.4.4 of the current Salem Updated Final Safety Analysis Report.
- The analysis methodology, postulated failures (i.e., Loss of offsite power (LOOP) assumed to occur concurrent with the reactor trip), duration of the event, and credited operator actions (PSE-96-579, Westinghouse letter "Evaluation of SGTR Analysis Using Assumed 30-Minute Operator Action Time" dated March 7, 1996) are also consistent with the current plant licensing basis.
- The input parameters for the thermal-hydraulic calculations were confirmed for the Salem Unit 2 replacement steam generators (PSE-05-10, Westinghouse letter "Replacement Steam Generator Project Request for Confirmation").

The results of the calculation are used for determination of the offsite and control room radiological consequences.

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

<u>TABLE 3.3-5</u>

<u>ENGINEERED SAFETY FEATURES RESPONSE TIMES</u>

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[illegible]
b. Reactor Trip (from SI)	≤2.0
c. Feedwater Isolation [illegible]	~~≤10.0~~ [illegible]
d. Containment Isolation-Phase "A"	≤17.0[illegible]/27.0[illegible]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤60
g. Service Water System	≤13.0[illegible]/45.0[illegible]
h. Containment Fan Coolers	≤60.0[illegible]

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

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

TABLE 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 [illegible]	~~≤ 10.0~~ [illegible]
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 [illegible] ~~[illegible]~~
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
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 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)	$\leq 27.0^{(1)}$
	b.	Reactor Trip (from SI)	≤ 2.0
	c.	Feedwater Isolation	~~≤ 10.0~~ [illegible]
	d.	Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
	e.	Containment Ventilation Isolation	Not Applicable
	f.	Auxiliary Feedwater Pumps	≤ 60
	g.	Service Water System	$\leq 13.0^{(2)}/45.0^{(3)}$
	h.	Containment Fan Coolers	$\leq 60.0^{(7)}$

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 [illegible]
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~~ [illegible]
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 [illegible]
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)	$\leq 12.0^{(2)}/22.0^{(3)}$
	b. Reactor Trip (from SI)	≤ 2.0
	c. Feedwater Isolation	~~≤ 10.0~~ Note 3
	d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
	e. Containment Ventilation Isolation	Not Applicable
	f. Auxiliary Feedwater Pumps	≤ 60
	g. Service Water System	$\leq 14.0^{(2)}/48.0^{(3)}$
	h. Steam Line Isolation	≤ 8.0
7.	Containment Pressure--High-High	
	a. Containment Spray	≤ 33.0
	b. Containment Isolation-Phase "B"	Not Applicable
	c. Steam Line Isolation	≤ 7.0
8.	Steam Generator Water Level--High-High	
	a. Turbine Trip	≤ 2.5
	b. Feedwater Isolation	≤ 10.0 Note 3
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	$\leq$ 60.0
11. Containment Radioactivity - High	
a. Purge and Pressure Vacuum Relief	$\leq$ 5.0(6)
12. Trip of Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
13. Undervoltage, Vital Bus	
a. Loss of Voltage	$\leq$ 4.0
14. Station Blackout	
a. Motor Driven Auxiliary Feed Pumps	$\leq$ 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 valve closure 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.

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~~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 3 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 ~~[illegible]~~ closure are credited in the containment analyses for LOCA and MSLB in case an FRV fails open.

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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.

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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.

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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.

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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.